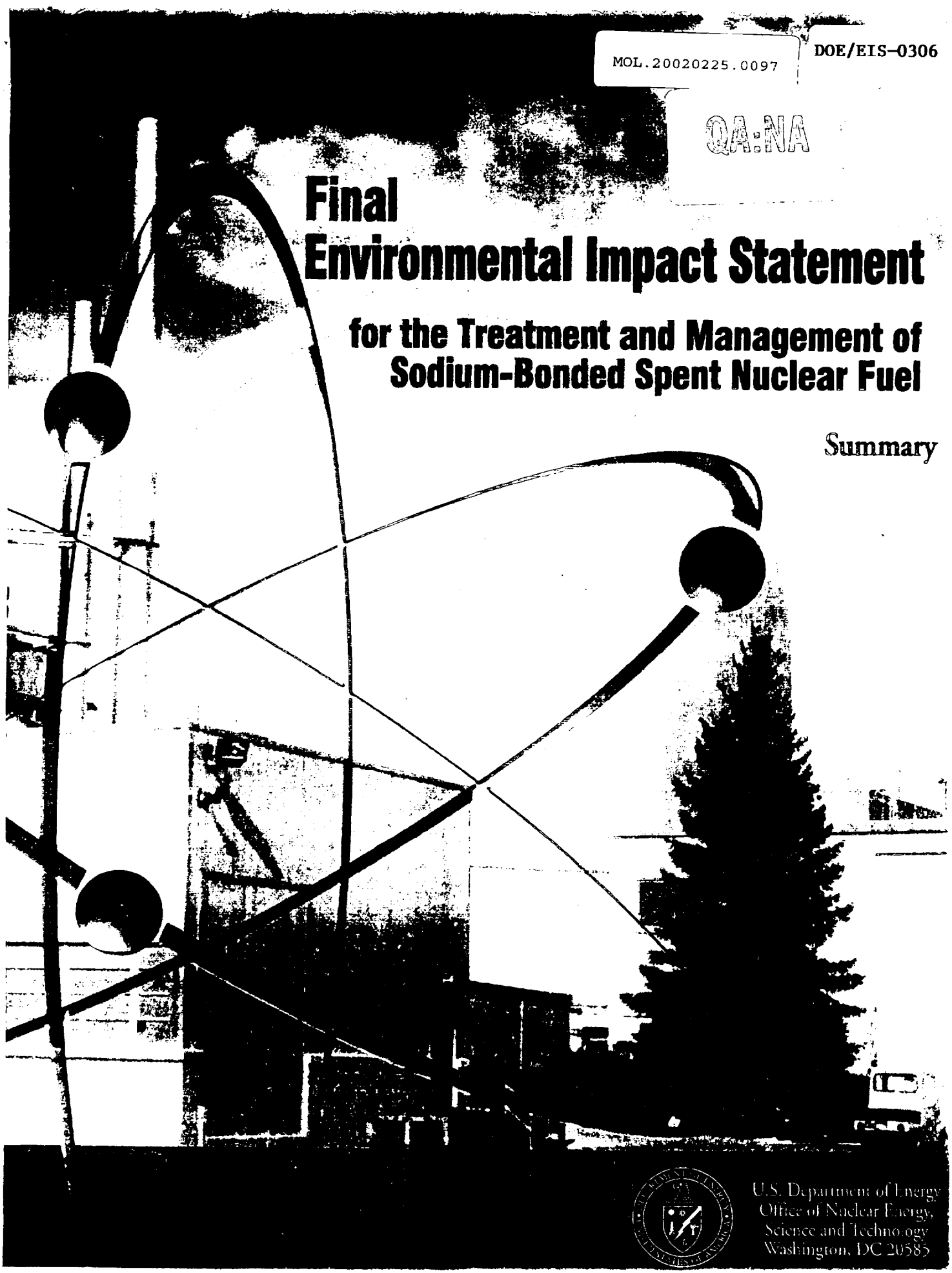
DOE/EIS-0306

Final Environmental Impact Statement

for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel

Summary



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COVER SHEET

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Abstract: DOE is responsible for the safe and efficient management of its sodium-bonded spent nuclear fuel. This fuel contains metallic sodium, a highly reactive material; metallic uranium, which is also reactive; and in some cases, highly enriched uranium. The presence of reactive materials could complicate the process of qualifying and licensing DOE's sodium-bonded spent nuclear fuel inventory for disposal in a geologic repository. Currently, more than 98 percent of this inventory is located at the Idaho National Engineering and Environmental Laboratory (INEEL), near Idaho Falls, Idaho. In addition, in a 1995 agreement with the State of Idaho, DOE committed to remove all spent nuclear fuel from Idaho by 2035. This EIS evaluates the potential environmental impacts associated with the treatment and management of sodium-bonded spent nuclear fuel in one or more facilities located at Argonne National Laboratory-West (ANL-W) at INEEL and either the F-Canyon or Building 105-L at the Savannah River Site (SRS) near Aiken, South Carolina. DOE has identified and assessed six proposed action alternatives in this EIS. These are: (1) electrometallurgical treatment of all fuel at ANL-W, (2) direct disposal of blanket fuel in high-integrity cans with the sodium removed at ANL-W, (3) plutonium-uranium extraction (PUREX) processing of blanket fuel at SRS, (4) melt and dilute processing of blanket fuel at ANL-W, (5) melt and dilute processing of blanket fuel at SRS, and (6) melt and dilute processing of all fuel at ANL-W. In addition, Alternatives 2 through 5 include the electrometallurgical treatment of driver fuel at ANL-W. Under the No Action Alternative, the EIS evaluates both the continued storage of sodium-bonded spent nuclear fuel until the development of a new treatment technology or direct disposal without treatment. Under all of the alternatives, the affected environment is primarily within 80 kilometers (50 miles) of spent nuclear fuel treatment facilities. Analyses indicate little difference in the environmental impacts among alternatives. DOE has identified electrometallurgical treatment as its Preferred Alternative for the treatment and management of all sodium-bonded spent nuclear fuel, except for the Fermi-1 blanket fuel. The No Action Alternative is preferred for the Fermi-1 blanket spent nuclear fuel.

Public Comments: The draft EIS was issued for public review and comment on July 31, 1999. The comment period ended on September 28, 1999, although late comments were accepted. Public hearings to solicit

comments on the draft EIS were held in North Augusta, South Carolina; Boise and Idaho Falls, Idaho; and Arlington, Virginia. All comments were considered during the preparation of the final EIS, which also incorporates additional and new information received since the issuance of the draft EIS. In response to comments on the SBSNF Draft EIS and as a result of information that was unavailable at the time of the issuance of the draft EIS, the final EIS contains revisions and new information. These revisions and new information are indicated by a double underline for minor word changes or by a sidebar in the margin for sentence or larger additions. Appendix A contains the comments received during the public review period of the SBSNF Draft EIS and DOE's responses to these comments. DOE will use the analyses presented in this final EIS as well as other information in preparing the Record of Decision for the treatment and management of its sodium-bonded spent nuclear fuel. DOE will issue this Record of Decision no sooner than 30 days after the U.S. Environmental Protection Agency publishes a notice of availability of this final EIS in the *Federal Register*.

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Acronyms and Abbreviations

ACRONYMS AND ABBREVIATIONS

ANL-W	Argonne National Laboratory-West
CFR	Code of Federal Regulations
CPP	Chemical Processing Plant
DOE	U.S. Department of Energy
EBR-II	Experimental Breeder Reactor-II
EIS	Environmental Impact Statement
FR	*Federal Register*
GMODS	Glass Material Oxidation and Dissolution System
INEEL	Idaho National Engineering and Environmental Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
MEDEC	melt, drain, evaporate, and calcine (ANL-W process)
NEPA	National Environmental Policy Act
NRC	U.S. Nuclear Regulatory Commission
PUREX	Plutonium-Uranium Extraction
RCRA	Resource Conservation and Recovery Act
SBSNF	Sodium-Bonded Spent Nuclear Fuel
SRS	Savannah River Site

SUMMARY

This document summarizes the U.S. Department of Energy's *Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel.* In addition to information concerning the background, purpose and need for the proposed action, and the National Environmental Policy Act process, this summary describes the characteristics of sodium-bonded spent nuclear fuel, the proposed treatment methods, the proposed facilities, the alternatives considered, and the environmental consequences of these alternatives. A glossary is included at the end to assist the reader with some of the technical terms used in this document.

S.1 BACKGROUND

The U.S. Department of Energy's (DOE) *Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel* (SBSNF EIS) identifies potential alternatives and impacts associated with the proposed treatment and management of DOE-owned sodium-bonded spent nuclear fuel and the facilitation of its disposal in a geologic repository. This environmental impact statement (EIS) was prepared in accordance with the National Environmental Policy Act (NEPA) of 1969, as amended.

For nearly four decades, research, development, and demonstration activities associated with liquid metal fast breeder reactors were conducted at the Experimental Breeder Reactor-II (EBR-II), about 40 miles west of Idaho Falls, Idaho; the Enrico Fermi Atomic Power Plant (Fermi-1) in Monroe, Michigan; and the Fast Flux Test Facility at the Hanford site in Richland, Washington. These activities generated approximately 60 metric tons of heavy metal of sodium-bonded spent nuclear fuel for which DOE is now responsible.

Sodium-bonded spent nuclear fuel is distinguished from commercial nuclear reactor spent nuclear fuel by the presence of metallic sodium, a highly reactive material; frequently by metallic uranium and plutonium, which also are potentially reactive; and in some cases, highly enriched uranium. Metallic sodium in particular presents challenges for management and ultimate disposal of this spent nuclear fuel. For example, metallic sodium reacts with water to produce explosive hydrogen gas and corrosive sodium hydroxide; both could affect operation of a geologic repository.

DOE proposes to treat and manage the sodium-bonded spent nuclear fuel and facilitate its ultimate disposal in a geologic repository. The reasonable alternatives for this proposed action are determined by the technology options available to DOE. Several technologies that might be used to treat and manage DOE's sodium-bonded spent nuclear fuel are at various stages of development. These include: (1) an electrometallurgical treatment process; (2) the plutonium-uranium extraction (PUREX) process; (3) placement of the spent nuclear fuel in high-integrity cans; (4) a melt and dilute process; (5) a glass material oxidation and dissolution system (GMODS) process; (6) a direct plasma arc-vitreous ceramic process; and (7) a chloride volatility process.

The programmatic risk in implementing any of these potential alternatives for treatment and management of sodium-bonded spent nuclear fuel, or of not treating this fuel, is the uncertainty surrounding the acceptability of DOE spent nuclear fuel for placement in a potential geologic repository. While DOE has drafted preliminary waste acceptance criteria[1], the final acceptance criteria would be more refined. If the proposed repository at Yucca Mountain in Nevada is developed, final acceptance criteria would not be available until about 2005, when the U.S. Nuclear Regulatory Commission (NRC) would issue a construction authorization. Until such time, the preliminary acceptance criteria are intended to be conservative to allow for uncertainties

[1] *Civilian Radioactive Waste Management System - Waste Acceptance System Requirements Document, 1999.*

in the performance of engineered and natural barriers and how such performance might impact public and worker health and safety, as well as material isolation.

This EIS follows the June 1, 1995, Record of Decision (60 FR 28680) for 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 Spent Nuclear Fuel EIS), in which DOE decided to regionalize the management of DOE-owned spent nuclear fuel by fuel type. DOE also decided to: (1) continue environmental restoration activities at the Idaho National Engineering and Environmental Laboratory (INEEL)[2]; (2) develop cost-effective treatment technologies for spent nuclear fuel and waste management; and (3) implement projects and facilities to prepare waste and treat spent nuclear fuel for interim storage and final disposition. The 1995 Record of Decision was based partially on the analyses in the Programmatic Spent Nuclear Fuel EIS, which analyzed the potential environmental consequences of alternatives for transporting, receiving, processing, and storing spent nuclear fuel under DOE's responsibility for the next 40 years. The Programmatic Spent Nuclear Fuel EIS also analyzed the consequences of 10 years of waste and spent nuclear fuel management and environmental restoration actions at INEEL.

In addition, DOE committed to remove all spent nuclear fuel from Idaho by 2035 in a 1995 agreement with the State of Idaho (Settlement Agreement and Consent Order issued on October 17, 1995, in the actions of *Public Service Co. of Colorado v. Batt*, No. CV 91-0035-S-EJL [D. Id.], and *United States v. Batt*, No. CV 91-0054-EJL [D. Id.]). Currently, more than 98 percent of DOE's sodium-bonded spent nuclear fuel is located at INEEL near Idaho Falls, Idaho, and is subject to the requirements of this Settlement Agreement and Consent Order. Before sodium-bonded spent nuclear fuel can be removed from the State of Idaho for ultimate disposal, some or all of the fuel may require treatment.

S.1.1 Purpose and Need for Action

Sodium-bonded spent nuclear fuel contains metallic sodium. The presence of metallic sodium in the sodium-bonded spent nuclear fuel could potentially complicate disposal certification and licensing for the ultimate disposal of this spent nuclear fuel in a geologic repository. Metallic sodium reacts vigorously with water, producing heat, potentially explosive hydrogen gas, and sodium hydroxide, a corrosive substance. Sodium also is pyrophoric (i.e., a material that is susceptible to spontaneous ignition and continuous combustion). Sodium metal was used as a heat-transfer medium within the stainless steel cladding (outer layer) of the nuclear fuel and as a coolant in the nuclear reactors that used this fuel. To the extent possible, sodium was removed from the external surface of this fuel after its use, but a portion remains bonded to the uranium metal alloy fuel within the cladding and cannot be removed without further treatment. Most (i.e., 99 percent by weight) of the sodium-bonded spent nuclear fuel contains metallic uranium and plutonium. Some metals, such as pure uranium and pure plutonium, are reactive in the presence of air and moisture. The current preliminary repository waste acceptance criteria exclude reactive and potentially explosive materials from being accepted into a geologic repository unless they exist only in trace quantities. Additionally, some of the sodium-bonded spent nuclear fuel contains highly enriched uranium that could create criticality concerns requiring control methods.

To ensure that the terms of the State of Idaho Settlement Agreement and Consent Order are met, and to facilitate disposal, DOE needs to reduce the uncertainties associated with qualifying sodium-bonded spent nuclear fuel for disposal. Appropriate treatment and management of the sodium-bonded spent nuclear fuel would significantly reduce complications related to disposal qualification. Technologies for spent nuclear fuel treatment that could facilitate such qualification therefore should be considered in reaching a decision for

[2] *The laboratory's name was changed from Idaho National Engineering Laboratory to Idaho National Engineering and Environmental Laboratory in January 1997.*

treatment of DOE-owned sodium-bonded spent nuclear fuel. Several treatment technologies are at various stages of development and could be used to remove and stabilize the metallic sodium and immobilize or isolate the transuranic and fission products that are in the sodium-bonded spent nuclear fuel. Such technologies include the electrometallurgical treatment process; the PUREX process; placement of the spent nuclear fuel in high-integrity cans; a melt and dilute process; the GMODS process; a direct plasma arc-vitreous ceramic process; and a chloride volatility process.

It is prudent to evaluate these alternative treatment technologies now, while DOE is performing site characterization activities for a potential geologic repository at Yucca Mountain, Nye County, Nevada. Potential waste forms resulting from treatment or packaging of sodium-bonded spent nuclear fuel should be developed as much as possible in parallel with any repository development. The process of establishing a repository depends on not only the site but also the materials for disposal. As part of this process, a total system performance assessment that describes the probable behavior of the repository is performed. This total system assessment includes the performance of the specific waste forms and inventories proposed for disposal. As part of the process of establishing a repository, data for the waste forms are needed prior to making a final repository selection.

Having completed the Electrometallurgical Treatment Research and Demonstration Project (see Section S.1.2.6.3) and in considering the future of PUREX processing capabilities, DOE now needs to decide whether these technologies are suitable for treating the remaining sodium-bonded spent nuclear fuel, or whether there is sufficient reason to delay a decision and wait for the development of other treatment technologies. Delaying this NEPA process could result in a loss of capability and of technical staff knowledgeable about and experienced with the demonstration project. This was an important consideration in the decision to proceed with this EIS.

Geologic Repository and Waste Acceptance Criteria

Geologic repositories are deep, excavated underground vaults constructed for the purpose of permanently containing nuclear waste. Any spent nuclear fuel packaging or treatment technology must be capable of putting fuel in a form that will satisfy acceptance criteria requirements. DOE has drafted preliminary acceptance criteria which are being used to assess the feasibility of DOE spent nuclear fuel disposition options. The draft criteria state that spent nuclear fuel containing materials that are explosive, pyrophoric, or chemically reactive in the repository environment would not meet the acceptance criteria. Because it contains metallic sodium, the sodium-bonded spent nuclear fuel could be categorized as hazardous waste that is potentially both pyrophoric and reactive.

These criteria would become more detailed, consistent with detailed designs of repository facilities and waste package performance. Under the schedule for the proposed geologic repository of Yucca Mountain in Nevada, final acceptance criteria would not be available until about 2005, when the NRC would issue a construction authorization. To ensure that the treatment option DOE selects will result in a product that is likely to meet the acceptance criteria, DOE is working with the NRC to obtain comments on the research and development work that DOE will perform to establish treatment technology specifications.

It is prudent to evaluate alternative treatment technologies now, while DOE is performing site characterization activities for a potential geologic repository at Yucca Mountain. Potential waste forms resulting from treatment or packaging of sodium-bonded spent nuclear fuel should be developed as much as possible in parallel with any repository development to promote consistency between the two efforts and to minimize programmatic risks associated with waste form qualification and acceptance for ultimate disposal.

Prior to this EIS, an environmental assessment was prepared for the Electrometallurgical Treatment Research and Demonstration Project at Argonne National Laboratory-West (ANL-W). A Finding of No Significant Impact was issued in May 1996. The finding required preparation of an EIS if electrometallurgical treatment would be proposed to treat the remaining EBR-II spent nuclear fuel, or if electrometallurgical treatment or another technology would be proposed on a production scale for the remaining sodium-bonded spent nuclear fuel owned by DOE. DOE is currently evaluating its options for the treatment and management of sodium-bonded spent nuclear fuel. A key element of DOE's decision-making is a thorough understanding of the environmental impacts that may occur during the implementation of the proposed action. NEPA provides Federal agency decision-makers with a process to consider potential environmental consequences (both positive and negative) of proposed actions before agencies make decisions. In following this process, DOE has prepared this EIS to assess various alternatives and to provide the necessary background, data, and analyses to help decision-makers and the public understand the potential environmental impacts of each alternative.

DOE's strategy for compliance with NEPA has been first to make decisions on programmatic alternatives in the Programmatic Spent Nuclear Fuel EIS, followed by site-specific analyses to implement the programmatic decisions.

Before an EIS can be prepared, the scope (i.e., the range of actions, alternatives, and impacts to be considered) must be determined. The NEPA process requires public participation in determining the scope of an EIS. The scoping process is initiated by a Federal agency's publication of a Notice of Intent to prepare an EIS in the *Federal Register*. DOE NEPA regulations require at least one public meeting and a minimum 30-day comment period to receive public input on the scope of the EIS.

S.1.2 NEPA Process

Following the completion of the scoping process, an agency issues a draft EIS for public review and comment. The public comment period on the draft EIS must be at least 45 days in duration, and under DOE's NEPA implementing procedures, at least one public meeting on the draft EIS must be held. DOE must consider all substantive comments on the draft EIS and address these comments in a final EIS. No sooner than 30 days after the notice of availability of the final EIS is issued by the EPA in the *Federal Register*, the agency may issue a Record of Decision; DOE publishes Records of Decision in the *Federal Register*.

This is the final EIS. The draft EIS was issued for public review and comment on July 31, 1999. The public comment period ended on September 28, 1999. In the final EIS, DOE responds to the public comments and, where appropriate, has made revisions to the document based on the public comments.

S.1.2.1 Issues Identified During the Scoping Period

On February 22, 1999, DOE published in the *Federal Register* a Notice of Intent to prepare an *Environmental Impact Statement for Electrometallurgical Treatment of Sodium-Bonded Spent Nuclear Fuel in the Fuel Conditioning Facility at Argonne National Laboratory-West* (64 FR 8553). In this Notice of Intent, DOE invited the public to participate and comment on the issues to be resolved in the EIS. Subsequent to this notice, DOE held four public scoping meetings. The first meeting was attended by about 60 persons and was held in Idaho Falls, Idaho, on March 9, 1999. The second meeting was held in Boise, Idaho, on March 11, 1999, and was attended by 7 persons. Ten persons attended the third meeting, which was held in North Augusta, South Carolina, on March 15, 1999. The fourth meeting was held in Arlington, Virginia, on March 18, 1999, and was attended by 8 persons. A court reporter recorded oral comments at each of these meetings. Written statements or comments from the public also were collected at the meetings. In addition, the public was invited to send comments to DOE by letter, electronic mail, a toll-free telephone number, and facsimile. The public scoping comment period began with the publication of the Notice of Intent in the *Federal Register* on February 22, 1999 (64 FR 8553), and ended 45 days later on April 8, 1999.

A total of 228 comments were received during the public scoping comment period. All comments were reviewed and considered by DOE in developing the scope of this EIS. A summary of scoping comments and their disposition is provided in Section A.1 of Appendix A of this EIS. The significant issues identified during the public scoping period are addressed below.

Many commentors at the public scoping meetings asked specific, technical questions about the proposed action. Areas of interest included:

- Waste volume reduction
- Nature of the spent nuclear fuel at ANL-W
- Waste forms characterization
- Waste disposition and qualification (repository acceptance criteria)
- PUREX process
- Use of facilities
- Nonproliferation impacts
- Transportation
- Demonstration project

A number of persons commented on the schedule for this EIS. Many stated that the draft EIS should not be issued for public comment before publication of other related reports, such as the National Research Council's Waste Qualification Assessment and the National Academy of Sciences' Independent Assessment Final Report on the Electrometallurgical Treatment Research and Demonstration Project; a Nonproliferation Impacts Assessment; and a Cost Study. Several commentors said that this EIS is premature because the demonstration project will not be completed until after the draft EIS is published.

Several commentors asked that the EIS include information about the costs of the proposed action and all of the technology alternatives under consideration. Other commentors stated that the public should have an

NEPA Definitions

NEPA of 1969, as amended: A law that requires Federal agencies to consider in their decision-making processes the potential environmental effects of proposed actions and analyses of alternatives and measures to avoid or minimize the adverse effects of a proposed action.

Alternatives: A range of reasonable options considered in selecting an approach to meeting the proposed objectives. In accordance with other applicable requirements, the No Action Alternative also is considered.

EIS: 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 decision-making, 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).

opportunity to comment on the independent Nonproliferation Impacts Assessment in the same time frame as the draft EIS, or that this EIS should be delayed until the Nonproliferation Impacts Assessment becomes publicly available. Some suggested that the Nonproliferation Impacts Assessment be included in the EIS. A few commentors expressed the opinion that electrometallurgical treatment of spent nuclear fuel is a proliferation-prone technology.

Many waste-related comments included opinions about whether low-enriched uranium, plutonium, noble metals, and other components of the waste stream should be viewed as waste or potentially valuable resources. Several commentors asked that the EIS clarify which specific waste forms would be generated by the treatment processes. Others said the EIS should clarify whether the waste would remain at the Savannah River Site (SRS) after processing or be returned to Idaho if the PUREX process were used. Some commentors argued that the electrometallurgical treatment alternative would not reduce the volume of waste to be stored in a repository. A few questioned how DOE can ensure the waste will meet the acceptance criteria for a repository when no one knows what those criteria will be—or if there will be any repository at all. A few others recommended that the EIS evaluate the PUREX process before it is shut down to ensure that the waste forms resulting from electrometallurgical treatment are as good as the borosilicate glass that is being prepared for a geologic repository.

The commentors generally agreed that DOE should evaluate in detail all of the alternative treatment technologies that potentially could meet DOE's treatment and management needs, even those that DOE considers less technologically mature. Several commentors expressed the opinion that DOE already has made a technology decision in favor of electrometallurgical treatment, but that other alternative new technologies should not be dismissed because of a lack of knowledge about them. Some asked that the EIS: (1) explain how DOE can consider the PUREX process a reasonable alternative when, historically, it could not handle sodium-bonded spent nuclear fuel, and (2) evaluate whether changes in the PUREX process would be needed to accommodate sodium-bonded spent nuclear fuel. A few commentors suggested the EIS should analyze blanket and driver spent nuclear fuel separately, since they have different chemical and radiological characteristics and different treatments might be warranted.

Comments concerning environment, safety, and health issues were comparatively few, as were comments about transportation safety and security.

Comments received during the scoping period were systematically reviewed and evaluated to determine whether the issues raised fell within the scope of the EIS. As a result of public comment, DOE changed the proposed action of the EIS, as well as the structure of the alternatives. The proposed action was changed from electrometallurgical treatment of sodium-bonded spent nuclear fuel in the Fuel Conditioning Facility at ANL-W to the treatment and management of sodium-bonded spent nuclear fuel. The title also was changed accordingly. This change was made to alleviate concerns about bias for one treatment technology over others. The alternatives were restructured to reflect differences in the characteristics of the sodium-bonded spent nuclear fuel types. Thus, several alternatives were added that treat blanket and driver spent nuclear fuel by different technologies.

Issues related to cost and nuclear nonproliferation were not considered to be within the scope of the EIS. However, DOE conducted a Cost Study and a Nonproliferation Impacts Assessment for the reasonable alternatives. These reports were made available to the public during the public review process.

With respect to comments related to the ongoing Electrometallurgical Treatment Research and Demonstration Project, data from the project were used for the preparation of both the draft and final EIS. The National Research Council issued a final report in April 2000 on the waste forms generated by the technology demonstration. DOE will consider the Council's final report in the Record of Decision process which follows the issuance of the final EIS.

S.1.2.2 Issues Raised During the Public Comment Period on the Draft EIS

In July 1999, DOE published the *Draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel.* The regulations implementing NEPA mandate a minimum 45-day public comment period after publication of a draft EIS to provide an opportunity for the public and other stakeholders to comment on the EIS analysis and results. The 45-day public comment period on the SBSNF draft EIS began on July 31, 1999, and was scheduled to end on September 13, 1999. In response to commentor requests, the comment period was extended an additional 15 days through September 28, 1999. During this 60-day comment period, public hearings were held in North Augusta, South Carolina; Boise and Idaho Falls, Idaho; and Arlington, Virginia. In addition, the public was encouraged to submit comments via the U.S. mail service, electronic mail, a toll-free 800-number phone line, and a toll-free fax line.

A total of 494 comments were received during the public comment period. Most of the comments focused on the following issues: (1) the purpose, need for, and timing of the proposed action; (2) the introduction of new waste forms produced by the proposed action, their acceptability in a geologic repository, and the disposition of uranium and plutonium by-products; (3) the public availability of information considered relevant to reviewing the draft EIS, the extension of the comment period, and the relationship of the EIS to other DOE programs; (4) the cost of the various alternatives; (5) the impacts of the proposed action on U.S. nuclear nonproliferation policy; (6) technical or NEPA-related questions regarding technologies and alternatives; and (7) questions related to the affected environment and the environmental consequences. DOE's responses to these issues are summarized below. The comments also dealt with a number of other subjects, including technologies considered and dismissed from further evaluation, long-term (beyond institutional control) performance of the sodium-bonded spent nuclear fuel during storage on site, and questions about the methodology and assumptions of the health and safety analysis. Many commentors expressed their opposition or support for DOE's action in general or for specific alternatives under the proposed action or the No Action Alternative. Volume 2, Section A.2 of Appendix A of this EIS provides the public hearings overview and DOE's responses to all comments on a comment-by-comment basis.

Purpose, Need for, and Timing of the Proposed Action

Many comments expressed the opinion that DOE failed to demonstrate the purpose and need for the proposed action or to provide a rationale for its timing. Some of the reasons given included the lack of a compelling argument that there is a safety risk associated with current storage; the lack of a regulatory framework and final waste acceptance criteria; the lack of an approved site for a geologic repository; insufficient information on the results of the Electrometallurgical Treatment Research and Demonstration Project; and the lack of analysis showing that direct disposal of the sodium-bonded spent nuclear fuel without sodium removal would be detrimental to the performance of the geologic repository.

DOE's position, presented in the EIS, is that the need to examine options for the treatment and management of sodium-bonded spent nuclear fuel is based on the existing regulatory environment concerning long-term disposal of spent nuclear fuel and high-level radioactive waste. DOE assumes that its sodium-bonded spent nuclear fuel, as well as other DOE-owned spent nuclear fuel, eventually will be disposed of in a geologic repository. However, one of the key requirements, as specified in the current April 1999 version of DOE's *Civilian Radioactive Waste Management System - Waste Acceptance Systems Requirements Document* and in NRC requirements for acceptance of spent nuclear fuel or high-level radioactive waste in a geologic repository, is that it cannot contain or generate materials that are explosive, pyrophoric, or chemically reactive in a form or amount that could compromise the repository's ability to perform its waste isolation function or to satisfy its performance objective [10 CFR 60.135(b)(1)]. The sodium-bonded spent nuclear fuel, if left in its existing state, would contain pyrophoric and chemically reactive metallic sodium and therefore may not meet DOE or NRC repository acceptance criteria, or would complicate the qualification process.

The timing for the proposed action is a programmatic issue rather than a safety issue. The EIS does not conclude that current storage of sodium-bonded spent nuclear fuel presents a threat to the health and safety of workers or the public. The programmatic risk associated with implementing the proposed action or not treating the sodium-bonded spent nuclear fuel is the uncertainty surrounding the acceptability of this fuel for placement in a geologic repository. The process of establishing a repository depends on not only the site but also the materials for disposal. As part of this process, a total system performance assessment that describes the probable behavior of a repository is performed. This total system assessment includes the performance of the specific waste forms and inventories proposed for disposal. As part of the process of establishing a repository, data for the waste forms are needed prior to making a final repository selection, not after. In fact, if specific waste forms are not represented in crucial documents like this EIS, additional documentation will be needed to allow the possibility of disposing of those materials in the repository. The performance of sodium-bonded spent nuclear fuel in a geologic repository depends on many factors (e.g., long-term fuel integrity and fuel/waste package survivability in a repository environment), and the presence of metallic sodium would complicate the modeling even further. Stabilization of the sodium-bonded spent nuclear fuel and/or removal of the metallic sodium would provide greater protection for human health and the environment.

The Electrometallurgical Treatment Research and Demonstration Project began in June 1996 and, although the review of the test results has not been finalized in a single report, a number of status reports were issued by DOE and reviewed by the National Academy of Sciences' National Research Council Committee. They are referenced in this EIS. The success criteria established at the outset of the project have been fulfilled. The environmental impact analysis associated with the electrometallurgical treatment process alternatives was based on actual data from the demonstration project. This final EIS includes a new section on the status and results of the project. Having completed the demonstration project and in considering the future of its PUREX processing capabilities, DOE now needs to decide whether these technologies are suitable for treating the remaining sodium-bonded spent nuclear fuel, or whether there is sufficient reason to delay a decision and wait for the development of other treatment technologies. Delaying the NEPA process could result in a loss of capability and of technical staff knowledgeable about and experienced with the demonstration project. This was an important consideration in the decision to proceed with this EIS.

New Waste Forms and Disposition of Uranium and Plutonium By-Products

Some of the comments questioned the generation of new waste forms from treating the sodium-bonded spent nuclear fuel and the possible acceptance of these forms in a geologic repository. Also, a number of commentors remarked on the generation of uranium and plutonium as by-products of the treatment process. Related issues were the disposition of uranium metal, a by-product of the electrometallurgical process, and the compliance of both the PUREX and the electrometallurgical processes with U.S. nuclear nonproliferation policy in terms of the separation of these elements.

All of the alternatives evaluated in this EIS would produce some form of high-level radioactive waste. Electrometallurgical treatment would produce two new waste forms (i.e., metallic and ceramic) and the melt and dilute process would produce a new metallic form (i.e., a melt and dilute product, or conditioned spent nuclear fuel). These forms would be more stable than the untreated sodium-bonded spent nuclear fuel. The production of a chemically stable waste form to replace a chemically reactive waste form (i.e., sodium-bonded spent nuclear fuel) represents an improvement in the safe, long-term storage of this spent nuclear fuel. DOE expects the new waste forms to be suitable for disposal in a repository and to meet the requirements of the final waste acceptance criteria. The high-level radioactive waste form resulting from the PUREX process is borosilicate glass, which has been tested and analyzed extensively under conditions relevant to a geologic repository.

With respect to uranium and plutonium disposition, the EIS states that any uranium that would be separated under the electrometallurgical process would be blended down and stored on site if it originates from driver

spent nuclear fuel, or would be stored on site as depleted uranium if it originates from blanket spent nuclear fuel. The final disposition of the stored uranium has not been decided and is not discussed in this EIS. The disposition of the uranium will be subject to a separate NEPA review. The nuclear nonproliferation policy aspects of this separation are subject to the nuclear nonproliferation policy assessment of the alternatives. The approximately 260 kilograms (572 pounds) of plutonium that would be separated under the PUREX process would be disposed of in accordance with the Record of Decision (75 FR 1608) for the *Surplus Plutonium Disposition Final Environmental Impact Statement* (DOE/EIS-0283) issued in November 1999. This separation is the subject of the Nonproliferation Impacts Assessment, which is independent of this EIS.

Public Availability of Information and Related Documentation

Many commentors asked for a 60-day extension of the 45-day public comment period on the draft EIS. Commentors said they wanted additional time to obtain and review relevant documents such as the Yucca Mountain Draft EIS and the National Academy of Sciences' National Research Council's final report on the Electrometallurgical Treatment Research and Demonstration Project, as well as the Cost Study and the Nonproliferation Impacts Assessment. The commentors frequently stated that DOE needs to make all of this information publicly available before the end of the EIS comment period and the issuance of the final EIS and the Record of Decision.

In an effort to ensure that all interested parties had time to comment on the draft EIS, the due date for transmittal of comments was extended from September 13 to September 28, 1999 (64 FR 49169). With respect to the need for more information, DOE made that information available to the public. Background materials were placed in public reading rooms and were made available to the public through a series of hearings held August 17, 1999, in North Augusta, South Carolina; August 24, 1999, in Boise, Idaho; August 26, 1999, in Idaho Falls, Idaho; and August 31, 1999, in Arlington, Virginia. Materials placed in the reading rooms included the electrometallurgical demonstration environmental assessment, the Finding of No Significant Impact for the environmental assessment, National Research Council reports, the 1995 Settlement Agreement and Consent Order with the State of Idaho, the scoping meeting transcripts and comments, and the draft EIS hearing presentations and fact sheets. In addition, completion of the Cost Study and Nonproliferation Impacts Assessment was expedited so that they would be available to the public during the comment period. These reports were mailed to interested parties on August 12, 1999, and were made available to attendees at all of the public hearings on the draft EIS. Although these reports are not critical to the evaluation of the analysis presented in the draft EIS, they will provide input to the Record of Decision. While the final National Research Council report on the demonstration project was published in April 2000, interim status reports were produced throughout the project. Data generated during the demonstration project were used in preparing the EIS.

Cost Issues

A number of commentors raised cost issues and provided comments directly related to the Cost Study, which is not part of the EIS.

Comments concerning the costs of the proposed action were considered beyond the scope of the EIS. The EIS was prepared in accordance with NEPA, as well as the Council on Environmental Quality's regulations on implementing NEPA (40 CFR 1500 through 1508) and DOE's NEPA regulations (10 CFR 1021). None of these regulations require the inclusion of a cost analysis in an EIS. The basic objective of the SBSNF EIS is to provide the public and DOE decision-makers with a description of the reasonable alternatives for treating and managing sodium-bonded spent nuclear fuel and information about their potential impacts on public health and safety and the environment. While cost could be an important factor in the ultimate Record of Decision, the purpose of this EIS is to address the environmental consequences of all alternatives under the proposed action and the No Action Alternative. DOE distributed cost information through the independent Cost Study

released in August 1999, and this information is available to the public on request and in the DOE's public reading rooms. Responses to specific comments related to cost issues are included in Volume 2, Section A.2 of Appendix A of the EIS.

Nuclear Nonproliferation Policy Issues

The nuclear nonproliferation implications of the proposed action were the subject of a number of comments. Some commentors expressed strong opinions about how the use of specific technologies such as electrometallurgical treatment might impact U.S. nonproliferation policy.

Nuclear Nonproliferation is another issue that was considered beyond the scope of the EIS. A separate Nonproliferation Impacts Assessment was prepared by DOE's Office of Arms Control and Nonproliferation. After assessing the potential nonproliferation impacts that could result from each of the alternatives and technologies analyzed in the SBSNF Draft EIS, the Office of Arms Control and Nonproliferation found that all the alternatives, except that involving PUREX processing at SRS, are fully consistent with U.S. policy concerning reprocessing and nuclear nonproliferation. Electrometallurgical treatment, for example, would not increase national inventories of weapons-usable fissile material because, although highly enriched uranium is an interim product of the process, it would be blended down to low-enriched uranium during treatment. Within the current equipment configuration and design, it is not possible to produce weapons-usable plutonium merely by adjusting the operating parameters. To do this, traditional aqueous processing would be required after electrometallurgical treatment. However, traditional aqueous processing could be used to produce weapons-usable plutonium directly from the spent nuclear fuel, without electrometallurgical treatment, so electrometallurgical treatment itself does not present a special proliferation concern. Responses to specific comments related to nuclear nonproliferation are included in Volume 2, Section A.2 of Appendix A of the EIS.

Technologies, Alternatives

Various comments dealt with technical questions and issues regarding the treatment technologies addressed in the EIS or NEPA-related issues regarding the selected alternatives.

The variety of the issues precludes a summary response. Responses to these questions on a comment-by-comment basis are included in Volume 2, Section A.2 of Appendix A of the EIS. A number of revisions to the EIS were made as a result of these comments.

Affected Environment and Consequences

A number of comments included questions concerning the description of the affected environment in the SBSNF Draft EIS, and the results of the environmental impact analysis.

As in the case above, responses to these questions on a comment-by-comment basis are included in Volume 2, Section A.2 of Appendix A of the EIS.

S.1.2.3 Changes from the Draft EIS

In response to comments on the SBSNF Draft EIS and as a result of information that was unavailable at the time of the draft EIS issuance, the final EIS contains revisions and new information. These revisions and new information are indicated by a double underline for minor word changes or by a sidebar in the margin for sentence or larger additions. Appendix A contains the comments received during the public review period of the SBSNF Draft EIS and DOE's responses to these comments. Responses to comments related to cost and nuclear nonproliferation issues, although included in the Appendix, did not result in any changes to the EIS. A brief discussion of the most important changes included in the final EIS, and reflected in the Summary, is provided in the following paragraphs.

Results of the Electrometallurgical Treatment Research and Demonstration Project

As a result of public concern that results of the demonstration project were not incorporated in the draft EIS, a section (Section 1.6.3) was added in the final EIS with a description, status, and results of the demonstration project (see also Section S.1.2.6.3).

Justification of Purpose and Need and Timing

As a result of public concern that the draft EIS did not adequately justify the need and timing for the proposed action, Section 1.2 of the final EIS was revised to reflect DOE's position and DOE's responses to the related comments (see also Section S.1.1).

Relationship to Other NEPA Actions

Section 1.6.2.2 of the final EIS was revised to update the information provided on the *Draft Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada,* which was issued in July 1999 (see also Section S.1.2.6.2). A discussion of the Surplus Plutonium Disposition Final EIS and Record of Decision was added to Section 1.6.1.8 of the final EIS (see also Section S.1.2.6.1).

Sodium Removal and Disposition

As a result of public comment, the description of an alternate method for decladding and cleaning sodium-bonded blanket spent nuclear fuel, the laser declad and alcohol wash process, was added in Section 2.3.9 of the final EIS (see also Section S.3.9). The reason the process was not included in the evaluation of the reasonable alternatives also is in these sections.

No Action Alternative Definition

One of the two options of the No Action Alternative was revised from "indefinite" storage until the development of a currently less mature technology to "continued storage of the sodium-bonded spent nuclear fuel until 2035 or until the development of a currently less mature technology." The revision clarifies the issue raised by public comments concerning the time period covered by this EIS. This EIS covers the time period until 2035.

In addition, under both options of the No Action Alternative, it was determined that the sodium-bonded spent nuclear fuel would be packaged at ANL-W in preparation for shipment out of the State of Idaho by 2035.

No Action Alternative Assumptions

As a result of public comment, the assumption for the calculation of the radiological gaseous emissions under the No Action Alternative was changed. The draft EIS conservatively assumed that the radiological gaseous emissions would be a fraction of the total radiological gaseous emissions presented in the Programmatic Spent Nuclear Fuel EIS, in direct proportion to the heavy-metal mass ratio of the sodium-bonded spent nuclear fuel to the total spent nuclear fuel stored at INEEL. The final EIS directly calculates the radiological gaseous emissions using a more realistic fuel degradation assumption based on historical evidence. This change considerably reduced the estimated radiological gaseous emissions as well as the resulting doses to workers and the public under the No Action Alternative.

Dose and Risk Calculations

As a result of public comments and the availability of recent data from the Electrometallurgical Treatment Research and Demonstration Project and the *Savannah River Site Spent Nuclear Fuel Management Final Environmental Impact Statement,* dose calculations were revised in the final EIS. These revisions include: (1) the addition of project total doses to workers, (2) the project total risk to the public under normal operations, and (3) changes in doses and risks to the public and workers from accidents. In addition, dose and risk values were rounded, resulting in some changes in the numerical values in the EIS.

Air Quality

Based on public comments on the draft EIS, concentrations and emissions from sources that operate in support of the processing alternatives at ANL-W (e.g., emergency generators) were quantified and added to Sections 3.2.3 and 3.3.3 (Air Quality and Noise) and Chapter 4 of the final EIS. In addition, the baseline nonradiological air quality concentrations for INEEL presented in the draft EIS were replaced with more current emission inventory data.

Land Use/Ecology

As a result of comments received on the draft EIS, reference to the newly established 29,950-hectare (74,000-acre) INEEL Sagebrush Steppe Ecosystem Reserve was added to Sections 3.2.1.1 (Land Use) and 3.2.6 (Ecological Resources) of the final EIS.

Water Quality

As a result of public comments, a discussion and a summary table of radioactive liquid effluent at both INEEL and SRS were added to Sections 3.2.4 and 3.3.4 (Water Resources) of the final EIS.

Geology and Soils

As a result of public comments on the draft EIS, material on earthquake activity and volcanism in the vicinity of INEEL (Section 3.2.5, Geology and Soils) was revised.

Existing Human Health Risk

As a result of public comments, baseline concentrations and associated hazard indexes or cancer risks for hazardous chemicals at both ANL-W and SRS were added to Sections 3.2.10 and 3.3.10 (Existing Human Health Risk) of the final EIS.

Waste Management

Records of Decision for the *Final Waste Management Programmatic Environmental Impact Statement for Managing Treatment, Storage, and Disposal of Radioactive and Hazardous Waste* addressing the management of high-level radioactive waste and low-level radioactive waste were issued on August 26, 1999 (64 FR 46661), and February 25, 2000 (65 FR 10061), respectively. A summary of these decisions was added to the waste management discussion for both INEEL and SRS (Sections 3.2.11 and 3.3.11, respectively, of the final EIS).

Cumulative Impacts

The cumulative impacts section (Section 4.11 of the final EIS) was updated to reflect recent information obtained from the *Idaho High-Level Waste and Facilities Disposition Draft Environmental Impact Statement* and the *Savannah River Site Spent Nuclear Fuel Management Final Environmental Impact Statement.*

Electric Energy Consumption

Section 4.14.3, Irreversible and Irretrievable Commitments of Resources, of the final EIS was revised to include electrical energy consumption associated with the alternatives under the proposed action.

Settlement Agreement and Consent Order with the State of Idaho

As a result of public comments concerning the contents of the Settlement Agreement and Consent Order with the State of Idaho, the entire document was included in the final EIS as Appendix K.

Melt and Dilute Driver Fuel at SRS

The option of using the melt and dilute process to treat sodium-bonded driver spent nuclear fuel at SRS was considered at the recommendation of a public comment. The option was dismissed from further evaluation as explained in the revised Section 2.6 of the final EIS.

Preferred Alternative

In accordance with requirements of the Council on Environmental Quality regulations (40 CFR 1502.14e), the final EIS incorporates DOE's Preferred Alternative for the treatment and management of sodium-bonded spent nuclear fuel. The Preferred Alternative is discussed in Section 2.8 (see also Section S.5.8).

Transportation

The analysis was expanded to include the impacts from transporting the various waste forms and spent nuclear fuel packages from ANL-W to the INEEL Dry Storage Facility prior to transporting materials out of the State of Idaho by 2035.

Miscellaneous Revisions and Editorial Changes

Several sections in the SBSNF Final EIS were revised to reflect the availability of more recent data or to include corrections, improvements in the presentation, and other editorial changes. Among these are a simplified presentation of the consequences (Section S.8) and the addition of cumulative impacts (Section S.9) in the Summary. None of these revisions affects the environmental impact analysis presented in the EIS.

S.1.2.4 Scope of this EIS

The EIS evaluates the potential direct, indirect, and cumulative environmental impacts associated with the treatment of sodium-bonded spent nuclear fuel in one or more spent nuclear fuel management facilities. In addition, this EIS evaluates the environmental impacts of the No Action Alternative.

DOE proposes to treat and manage sodium-bonded spent nuclear fuel at one or more of the following spent nuclear fuel management facilities: ANL-W at INEEL and the F-Canyon or Building 105-L at SRS. The impacts from the treatment and management of sodium-bonded spent nuclear fuel at INEEL and SRS and their spent nuclear fuel management facilities are described in this EIS. In addition to the No Action Alternative, the EIS analyzes six reasonable alternatives under the proposed action that employ one or more of the following technology options: electrometallurgical treatment, the PUREX process, packaging in high-integrity cans, and the melt and dilute treatment process. Electrometallurgical treatment at a site other than ANL-W, the GMODS process, the direct plasma arc-vitreous ceramic treatment, and the chloride volatility process were considered and deemed not to be reasonable alternatives under the proposed action.

This EIS analyzes the potential environmental impacts associated with the proposed action, which includes: (1) preparation prior to treatment; (2) treatment and management; (3) transportation; and (4) decontamination and deactivation of equipment that would be installed for the purpose of implementing a specific treatment method. Impacts from the transport to INEEL of sodium-bonded spent nuclear fuel from DOE sites such as the Hanford site in Washington, Sandia National Laboratories in New Mexico, and Oak Ridge National Laboratory in Tennessee are addressed in the Programmatic Spent Nuclear Fuel EIS.

The United States does not encourage the civilian use of plutonium and, accordingly, does not itself engage in plutonium reprocessing for either nuclear power or nuclear explosive purposes. However, two of the technologies under the proposed action involve the separation of plutonium (PUREX) and highly enriched uranium (electrometallurgical treatment). To address concerns that treatment of this fuel by chemical separation could encourage reprocessing in other countries, DOE's Office of Nonproliferation and National Security independently evaluated the impacts of each treatment technology on U.S. nonproliferation efforts. The Nonproliferation Impacts Assessment was published at about the same time as the draft EIS.

S.1.2.5 Decisions to be Made

Based on the analytical results of this EIS as well as cost, schedule, and nonproliferation considerations, DOE intends to make the following decisions:

- Whether to use an existing, mature technology to treat the sodium-bonded spent nuclear fuel, and if so, which technology should be selected and where should it be implemented.

- Whether to take no action now and wait for further information regarding the potential development of a geologic repository, or promote the development of a less mature (e.g., GMODS, plasma arc) or new treatment technology.

The information presented in this EIS, combined with public comments on the draft EIS, the Nonproliferation Impacts Assessment, the Cost Study of the reasonable alternatives, and the National Research Council's final evaluation of the demonstration project, will enable DOE to make a decision regarding treatment and management of the sodium-bonded spent nuclear fuel. DOE could make a different decision for each type of sodium-bonded spent nuclear fuel.

S.1.2.6 Relationship to Other Actions and Programs

This section explains the relationship between this EIS and other relevant NEPA documents. Completed NEPA actions are described in Section S.1.2.6.1, ongoing actions are described in Section S.1.2.6.2, and the Electrometallurgical Treatment Research and Demonstration Project in Section S.1.2.6.3.

S.1.2.6.1 Completed NEPA Actions

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, April 1995)

The Programmatic Spent Nuclear Fuel EIS analyzed at a programmatic level the potential environmental consequences of alternatives used for 40 years to transport, receive, process, and store spent nuclear fuel under DOE's responsibility. It also analyzed the consequences of 10 years of waste and spent nuclear fuel management and environmental restoration actions at the Idaho National Engineering Laboratory (now known as INEEL). For programmatic spent nuclear fuel management, this document analyzed alternatives that included no action, decentralization, regionalization, centralization, and the use of plans that existed in 1992 and 1993 for the management of these materials. For INEEL, this document analyzed alternatives such as no action, a 10-year plan, and minimum and maximum treatment, storage, and disposal of DOE waste. The SBSNF EIS was prepared as a follow-on to this programmatic EIS.

Savannah River Site Waste Management Final Environmental Impact Statement (DOE/EIS-0217, October 1995)

DOE issued this EIS to provide a basis for the selection of a site-wide approach to managing present and future (through 2024) waste generated at SRS. This waste would come from ongoing operations and potential actions, new missions, environmental restoration, and decontamination and decommissioning programs. The SRS Waste Management EIS is relevant to the SBSNF EIS because it evaluates management alternatives for various types of waste that actions proposed in this SBSNF EIS could generate.

Final Environmental Impact Statement, Interim Management of Nuclear Materials (DOE/EIS-0220, December 1995)

In this EIS, DOE evaluated actions to stabilize nuclear materials at SRS that present potential environmental, safety, and health risks in their current storage condition or may present a risk within the next 10 years. This Interim Management EIS evaluates treatment and management alternatives for spent nuclear fuel and other waste materials at SRS such as those generated by the proposed actions in the SBSNF EIS.

Environmental Assessment for the Electrometallurgical Treatment Research and Demonstration Project in the Fuel Conditioning Facility at Argonne National Laboratory-West (DOE/EA-1148, May 1996)

This NEPA analysis addressed the environmental impacts associated with a research and demonstration project involving the electrometallurgical treatment of up to 100 EBR-II driver spent nuclear fuel assemblies and up to 25 EBR-II blanket spent nuclear fuel assemblies in the Fuel Conditioning Facility at ANL-W. As noted in the environmental assessment, DOE had identified electrometallurgical treatment as a promising technology to treat EBR-II spent nuclear fuel, but an appropriate demonstration was needed to provide DOE with sufficient information to evaluate its technical feasibility. The successful demonstration of the electrometallurgical treatment technology on EBR-II spent nuclear fuel, combined with research and testing of the resulting waste forms, provides DOE with the information needed to determine whether this treatment

technology should be used to treat the remainder of EBR-II spent nuclear fuel and/or other types of spent nuclear fuel. The demonstration project is discussed in Section S.1.2.6.3.

Disposition of Surplus Highly Enriched Uranium Final Environmental Impact Statement (DOE/EIS-0240, June 1996)

DOE prepared this EIS because of the need to move rapidly to neutralize the proliferation threat of surplus highly enriched uranium and to demonstrate to other nations the United States' commitment to nonproliferation. The Disposition of Surplus Highly Enriched Uranium EIS evaluates the disposition and management alternatives for materials that actions proposed in this SBSNF EIS could generate.

Final Waste Management Programmatic Environmental Impact Statement for Managing Treatment, Storage, and Disposal of Radioactive and Hazardous Waste (DOE/EIS-0200, May 1997)

The Waste Management Programmatic EIS examined the potential environmental and cost impacts of strategic management alternatives for managing five types of radioactive and hazardous waste that have resulted and will continue to result from nuclear defense and research activities at a variety of sites around the United States. The five waste types are mixed waste, low-level radioactive waste, transuranic waste, high-level radioactive waste, and hazardous waste. The Waste Management Programmatic EIS provided information on the impacts of various siting alternatives which DOE will use to decide at which sites to locate additional treatment, storage, and disposal capacity for each waste type. This Programmatic EIS evaluates management and treatment alternatives for various types of waste material that actions proposed in the SBSNF EIS could generate.

Advanced Mixed Waste Treatment Project Final Environmental Impact Statement (DOE/EIS-0290, January 1999)

The Advanced Mixed Waste EIS assessed the potential environmental impacts associated with four alternatives related to the construction and operation of the Advanced Mixed Waste Treatment Facility at INEEL. The Advanced Mixed Waste Treatment Facility will treat transuranic waste, mixed waste, and alpha-contaminated mixed waste, at INEEL in preparation for disposal. After treatment, transuranic waste would be disposed of at the Waste Isolation Pilot Plant in New Mexico. Mixed waste would be disposed of at an approved disposal facility based on DOE's Waste Management Programmatic EIS.

Surplus Plutonium Disposition Final Environmental Impact Statement (DOE/EIS-0283, November 1999)

The Surplus Plutonium Disposition EIS evaluated reasonable alternatives for the siting, construction, and operation of facilities required to implement DOE's disposition strategy for surplus plutonium. The facilities analyzed include a pit disassembly and conversion facility, a plutonium conversion and immobilization facility, and a mixed oxide fuel fabrication facility. The Surplus Plutonium Disposition EIS also analyzed the potential impacts of fabricating a limited number of mixed oxide fuel assemblies for testing in a reactor. DOE selected SRS as the location for all of these facilities. The Surplus Plutonium Disposition EIS addresses the disposition of material that the SBSNF EIS could generate.

S.1.2.6.2 Ongoing NEPA Actions

Savannah River Site Spent Nuclear Fuel Management Final Environmental Impact Statement (DOE/EIS-0279, March 2000)

The SRS Spent Nuclear Fuel Management EIS analyzed the potential impacts from the safe and efficient management of spent nuclear fuel and targets assigned to SRS, including placing these materials in forms

suitable for ultimate disposition. Options to treat, package, and store spent nuclear fuel are discussed in this document. The alternatives considered in the SRS Spent Nuclear Fuel Management EIS encompass a range of new packaging, new processing, and conventional processing technologies for the treatment of spent nuclear fuel. Most of these processing technologies are also analyzed in this SBSNF EIS. The preferred alternative identified in the SRS Spent Nuclear Fuel Final Management EIS would prepare nearly all of the spent nuclear fuel at SRS for disposition using a melt and dilute treatment process. The remaining material would be managed using chemical separation. The SRS Spent Nuclear Fuel Management EIS evaluates management and treatment alternatives for spent nuclear fuel and other waste materials that actions proposed in the SBSNF EIS could process and generate.

Draft Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada (DOE/EIS-0250D, July 1999)

The proposed action addressed in this EIS is to construct, operate and monitor, and eventually close a geologic repository at Yucca Mountain in southern Nevada for the disposal of spent nuclear fuel and high-level radioactive waste currently in storage at 72 commercial and 5 DOE sites across the United States. The EIS evaluates (1) projected impacts on the Yucca Mountain environment from the construction, operation and monitoring, and eventual closure of the geologic repository; (2) the potential long-term impacts of repository disposal of spent nuclear fuel and high-level radioactive waste; (3) the potential impacts of transporting these materials nationally and in the State of Nevada; and (4) the potential impacts of not proceeding with the proposed action. Included in the high-level radioactive waste that is assumed to be disposed of at the repository are the metallic and ceramic waste forms that would be produced by the electrometallurgical treatment of both driver and blanket sodium-bonded spent nuclear fuel.

Under the No Action Alternative, this EIS evaluates the potential impacts of the continued storage of spent nuclear fuel and high-level radioactive waste at the current storage locations using two scenarios: the first assumes continued storage under institutional controls for at least 10,000 years, and the second assumes no institutional controls after 100 years. This EIS also evaluates the disposal at the proposed repository of spent nuclear fuel or high-level radioactive waste that may be generated by the proposed action presented in the SBSNF EIS.

Idaho High-Level Waste and Facilities Disposition Draft Environmental Impact Statement

This draft EIS was issued in December 1999. It evaluates alternatives for managing the high-level radioactive waste and associated radioactive waste and facilities at INEEL. Under the terms of the 1995 Settlement Agreement and Consent Order with the State of Idaho, DOE agreed to treat high-level radioactive waste currently stored at INEEL and to prepare the waste in a form ready to be shipped out of the State of Idaho by 2035. The purpose of this EIS is to assist DOE in making decisions concerning the management of this radioactive waste to ensure compliance with applicable laws and regulations, and protect the environment and the health and safety of the workers and the public in a cost-effective manner. The proposed action under this EIS contributes to the cumulative impacts at the site discussed in the SBSNF EIS.

S.1.2.6.3 Electrometallurgical Treatment Research and Demonstration Project

Before electrometallurgical treatment could be considered as a technology choice for treating EBR-II spent nuclear fuel, an appropriate demonstration project was needed to evaluate its technical feasibility. As a preliminary step to demonstration, DOE requested that the National Academy of Sciences' National Research Council conduct an independent assessment of electrometallurgical treatment technology and its potential application to EBR-II spent nuclear fuel. In its report, published in 1995, the National Research Council recommended DOE proceed with demonstrating the technical feasibility of electrometallurgical treatment using a fraction of the EBR-II spent nuclear fuel. Following the National Research Council's recommendation, DOE

conducted an environmental assessment of the demonstration project. The environmental assessment was completed in May 1996 and resulted in a Finding of No Significant Impact, so that no further NEPA review was necessary for the demonstration project to proceed.

In June 1996, DOE initiated a three-year testing program at ANL-W to demonstrate the technical feasibility of electrometallurgical treatment of up to 100 EBR-II driver spent nuclear fuel assemblies and up to 25 depleted uranium EBR-II blanket spent nuclear fuel assemblies. These two types of EBR-II spent nuclear fuel, driver and blanket, are typical of most of DOE's sodium-bonded spent nuclear fuel inventory. The number of driver spent nuclear fuel assemblies was selected to provide the minimum fission product loading (approximately 3 percent) needed to evaluate the effectiveness of the removal of fission products from the electrorefiner salt and their concentration in the ceramic waste form. The blanket spent nuclear fuel assemblies were treated using a high-throughput electrorefiner that was installed in ANL-W's Fuel Conditioning Facility to evaluate higher-efficiency electrorefining.

A total of 100 driver spent nuclear fuel assemblies were treated. These assemblies required multiple batch operations of the treatment equipment in a remote, radioactive hot cell with an inert argon atmosphere. These operations were considered sufficient to demonstrate a dependable, predictable process, including uptime, repair and maintenance, and the operability of the linked process steps. A repeatability demonstration was completed by processing 12 driver spent nuclear fuel assemblies under the same processing conditions. In addition, processing 100 driver spent nuclear fuel assemblies dissolved sufficient active fission products in the electrorefiner salt so that ceramic waste form samples could be produced with representative waste loadings. The purpose of including blanket spent nuclear fuel assemblies in the test program was to demonstrate the mass throughput capacity of the process equipment and facility. A one-month throughput test was completed and a total of 13 blanket spent nuclear fuel assemblies were treated by the end of August 1999, when the demonstration project was concluded.

To support the Electrometallurgical Treatment Research and Demonstration Project, DOE established an extensive research and development program at Argonne National Laboratory-East. The largest element of this research and development program involved development, testing, and qualification of the ceramic waste form. Another element was experimental support for electrorefining and metal processing operations in the Fuel Conditioning Facility. In addition, the research and development program included a modeling activity aimed at understanding and improving the electrometallurgical treatment process as well as laying out the requirements for production-scale treatment of the remaining EBR-II spent nuclear fuel. The combined results of the research and development program at Argonne National Laboratory-East and the spent nuclear fuel treatment operations at ANL-W provided the technical basis for final evaluation of the electrometallurgical treatment process. An extensive series of topical reports was prepared to present the results of the demonstration in detail. These reports were the basis for ANL-W's summary report on the demonstration project.

To assist in monitoring the progress of the demonstration project, DOE requested that the National Research Council establish a review committee, the Committee on Electrometallurgical Techniques for DOE Spent Fuel Treatment, to evaluate the technology and its development. Working with DOE and the National Research Council committee, ANL-W established four criteria for evaluating the demonstration. The evaluation criteria for the electrometallurgical spent fuel demonstration project are listed below.

Criterion 1: *Demonstrate that 100 driver and up to 25 blanket EBR-II assemblies can be treated in a Fuel Conditioning Facility within three years, with a throughput rate of 16 kilograms per month for driver assemblies sustained for a minimum of three months, and a blanket spent nuclear fuel throughput rate of 150 kilograms per month sustained for one month.*

Criterion 2: *Quantification (for both composition and mass) of recycle, waste, and product streams that demonstrate projected material balance with no significant deviations.*

Criterion 3: *Demonstrate an overall dependable and predictable process considering uptime, repair and maintenance, and operability of the linked process steps.*

Criterion 4: *Demonstrate that safety risks, environmental impacts, and nuclear materials accountancy are quantified and acceptable within regulatory limits.*

Based on a comparison of the demonstration results with the above criteria for success, the demonstration project was a technical success. All key performance criteria were met or exceeded. The results of the demonstration project proved the technical feasibility of using electrometallurgical treatment technology to process DOE's inventory of sodium-bonded spent nuclear fuel. In addition, the demonstration project validated the throughput rate of the sodium-bonded spent nuclear fuel, quantified all process streams, fine-tuned the operational parameters, refined the electrometallurgical treatment equipment, and provided actual waste forms for characterization. This last accomplishment was of particular importance because, as the Defense Waste Vitrification Project at SRS has shown, waste characterization is a lengthy process. Waste forms must be subjected to detailed chemical analysis and long periods of exposure to expected repository conditions. The waste form characterization in the electrometallurgical treatment demonstration project has already initiated the waste acceptance process. Preliminary results of waste form testing indicate that both the metallic and ceramic waste forms produced by the electrometallurgical process appear to be comparable to borosilicate glass, which has been tested and analyzed extensively under conditions relevant to a geologic repository.

The review committee of the National Academy of Sciences' National Research Council has continuously reviewed the progress of the Electrometallurgical Treatment Research and Demonstration Project and all reports to date have found the process to be proven for treating sodium-bonded spent nuclear fuel.

In the most recent status report issued in the summer of 1999, the National Research Council Committee expressed some concerns about the long-term performance and potential releases from the waste forms under repository conditions. However, as noted above, work completed at ANL-W since the latest National Research Council review of the project indicates that both the ceramic and metallic electrometallurgical treatment waste forms appear to be comparable to borosilicate glass, which has been tested and analyzed extensively under conditions relevant to a geologic repository. A final report from the National Research Council Committee was published in April 2000. The National Research Council's final report on *Electrometallurgical Techniques for DOE Spent Fuel Treatment* concluded that "The EBR-II demonstration project has shown that the electrometallurgical technique can be used to treat sodium-bonded spent nuclear fuel." The report further stated that "The committee has found no significant technical barriers in the use of electrometallurgical technology to treat EBR-II spent fuel, and EMT therefore represents a potentially viable technology for DOE spent nuclear fuel treatment." DOE will consider the Council's final report during the Record of Decision process which follows the issuance of the final EIS.

S.2 Sodium-Bonded Spent Nuclear Fuel Characteristics

As a result of research, development, and demonstration activities associated with liquid metal fast breeder reactors, DOE has approximately 60 metric tons of heavy metal of sodium-bonded spent nuclear fuel in its inventory. This represents approximately 2 percent of DOE's total current spent nuclear fuel inventory of nearly 2,500 metric tons of heavy metal.

The bulk of the sodium-bonded spent nuclear fuel in DOE's inventory is of two general types: driver fuel and blanket fuel. Driver fuel is used mainly in the center of the reactor core to "drive" and sustain the fission chain

reaction. It is highly enriched in the fissile isotope uranium-235. Blanket fuel is made from depleted uranium, a type of uranium in which most of the fissile uranium-235 has been removed. Blanket fuel is usually placed at the outer perimeter of the core and is used to breed plutonium-239, a fissile material. Blanket fuel contains primarily the nonfissile isotope uranium-238, which converts to fissile plutonium-239 as it absorbs the neutrons produced from the fission process. Typical blanket and driver spent nuclear fuel elements are shown schematically in **Figure S–1**.

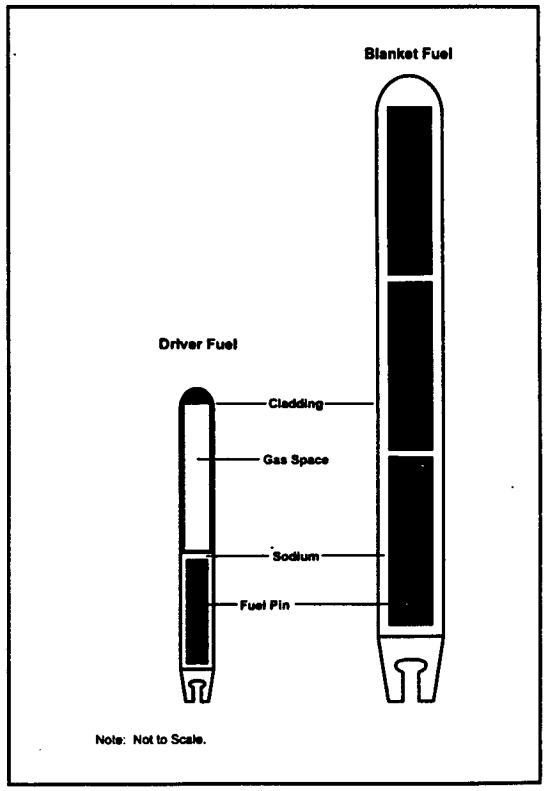


Figure S–1 Typical Driver and Blanket Spent Nuclear Fuel Elements

The blanket and driver spent nuclear fuel addressed by this EIS contain metallic sodium between the cladding (outer layer of the fuel element) and the metallic fuel pins to improve heat transfer from the fuel to the reactor coolant through the stainless steel cladding. When driver fuel is irradiated for some period of time, the metallic fuel swells as fission products are generated until it reaches the cladding wall. During this process, metallic sodium enters the metallic fuel and becomes inseparable from it. In addition, fuel and cladding components interdiffuse to such an extent that mechanical stripping of the driver spent nuclear fuel cladding is not practical. On the other hand, when blanket fuel is irradiated, the metallic fuel does not swell to the same degree as the driver fuel because the burnup in the blanket fuel is low. As a result, minimal metallic sodium enters the fuel pin and there is no interdiffusion between the fuel and cladding. This allows mechanical stripping of the blanket spent nuclear fuel cladding.

S.2.1 EBR-II Spent Nuclear Fuel

EBR-II driver spent nuclear fuel is stainless steel-clad, highly enriched uranium in a uranium alloy, typically either zirconium or fissium (an alloy of molybdenum, ruthenium, rhodium, palladium, zirconium, and niobium). The typical EBR-II driver spent nuclear fuel pin is a metal alloy of 90 percent uranium and 10 percent zirconium, or 95 percent uranium and 5 percent fissium. This fuel pin and a small amount of metallic sodium are loaded into a 74-centimeter-long (29-inch-long) stainless steel tube (cladding) and welded shut. This unit of fuel is called an element. Sixty-one (91 for some fuel) fuel elements are put together in a stainless steel hexagonal duct to make a fuel assembly approximately 2.3 meters (92 inches) long and 5.8 centimeters (2.3 inches) across.

Sodium-Bonded Spent Nuclear Fuel

Driver Fuel: During irradiation, the fuel expands. Metallic sodium enters the fuel and becomes mechanically inseparable from the uranium. In addition, the fuel and cladding mix so that mechanical removal of the cladding is not practical.

Blanket Fuel: When irradiated, only minimal amounts of sodium enter the fuel, and there is little or no mixing of the fuel and the cladding. Blanket fuel could be declad and the sodium removed.

Because of these differences between irradiated driver fuel and blanket fuel, the alternatives propose different treatment methods for each fuel type.

The EBR-II blanket spent nuclear fuel consists of stainless steel-clad, depleted uranium in metallic form. In EBR-II, the blanket spent nuclear fuel assemblies were used primarily for shielding and for reducing the required size of the reactor core. Blanket assemblies were placed outside of a stainless steel shield for all but the first few years of EBR-II operation. Blanket assemblies are similar to driver assemblies, except the blanket pins are made entirely from depleted uranium and the individual blanket pins are larger.

S.2.2 Fermi-1 Spent Nuclear Fuel

The Fermi-1 blanket spent nuclear fuel consists of stainless steel-clad, depleted uranium in a uranium-molybdenum alloy. Fermi-1 blanket elements are similar to EBR-II blanket elements in enrichment, but differ in dimensions (Fermi-1 elements are larger), form (uranium-molybdenum alloy versus uranium metal), and burnup. Because of its lower burnup, the Fermi-1 blanket fuel, which contains only about 0.2 percent plutonium by weight compared to approximately 1 percent plutonium by weight for the EBR-II blanket fuel, is subject to less stringent safeguard and security requirements than the EBR-II blanket fuel. This is an important consideration in the cost of managing the storage of these two fuel types.

After the Fermi-1 reactor was permanently shut down, the blanket assemblies were placed into 14 canisters and transported to the Idaho Nuclear Technology and Engineering Center (INTEC)—formerly known as the Idaho Chemical Processing Plant (CPP)—in 1974 and 1975 in 14 shipments. The canisters, made of stainless steel with a carbon steel basket inside, were placed into CPP-749, an underground dry storage system. The blanket assemblies have a very low irradiation history, so the inventory of fission products, activation products, and transuranics is low.

S.2.3 Fast Flux Test Facility and Miscellaneous Sodium-Bonded Spent Nuclear Fuel

DOE's inventory of sodium-bonded spent nuclear fuel includes eight liquid metal reactor test assemblies containing driver fuel that were irradiated at the Fast Flux Test Facility in Hanford. It also includes small quantities of fuel from liquid metal reactor experiments that have metallic sodium or an alloy of sodium and potassium. These miscellaneous small-lot fuels differ in cladding composition, uranium content, enrichment, and burnup. Some of the fuel consists of uranium and/or plutonium carbides, nitrides, and oxides in addition

Sources of Sodium-Bonded Spent Nuclear Fuel

EBR-II is a 62.5-megawatts-thermal research and test reactor located at ANL-W that was used to demonstrate the engineering feasibility of a sodium-cooled, liquid metal fast breeder reactor with a steam electric power plant and integral fuel cycle. It achieved initial criticality in September 1961 and continued to operate until September 1994. During its 33 years of operation, numerous fuel designs were tested in EBR-II, and EBR-II spent nuclear fuel contains both driver and blanket fuel.

Fermi-1 was built at Monroe, Michigan (30 miles southwest of Detroit), to demonstrate the feasibility of the fast breeder reactor for electric power production. Fermi-1 was a sodium-cooled, fast reactor. The reactor achieved initial criticality in 1963 and operated until September 1972. Fermi-1 was licensed for operation at a power level of 200 megawatts-thermal. Only blanket spent nuclear fuel from Fermi-1 is sodium-bonded.

The **Fast Flux Test Facility Reactor**, located on the Hanford site near Richland, Washington, is a 400-megawatts-thermal nuclear test reactor cooled by liquid sodium. It was built in 1978 to test plant equipment and fuel for the U.S. Government's liquid metal reactor development program. Although the Fast Flux Test Facility Reactor is not a breeder reactor, this program demonstrated the technology of commercial breeder reactors. The sodium-bonded spent nuclear fuel from the Fast Flux Test Facility Reactor is experimental fuel.

to metal uranium or alloy. For the purposes of this EIS, this miscellaneous fuel is assumed to have driver fuel characteristics. This fuel is stored at several DOE sites such as the Hanford Site, Oak Ridge National Laboratory, SRS, Sandia National Laboratories/New Mexico, and INEEL. Those lots stored outside INEEL will be transported to INEEL pursuant to the Record of Decision (60 FR 28680) for the Programmatic Spent Nuclear Fuel EIS.

Table S–1 provides a summary of the spent nuclear fuel addressed by this EIS. As described earlier, the majority of the spent nuclear fuel consists of EBR-II driver fuel, EBR-II blanket fuel, and Fermi-1 blanket fuel.

Table S–1 Overview of Sodium-Bonded Spent Nuclear Fuel Categories

Spent Nuclear Fuel Type	*Storage Volume* [a] *(cubic meters)*	*Metric Tons of Heavy Metal*	*Sodium Content (kilograms)*
EBR-II driver	58 [b]	3	83
EBR-II blanket	13	22	176
Fermi-1 blanket	19	34	365
Fast Flux Test Facility driver	8 [b]	0.3	7
Miscellaneous [c]	3 [b]	0.1	31
Total	101	60	662

[a] Volume refers to the canister storage volume.
[b] A larger volume per unit mass is required for driver spent nuclear fuel for criticality control.
[c] Assumed to have driver fuel characteristics.

Table S–2 provides information on the DOE sites where the sodium-bonded spent nuclear fuel is being stored, the locations within each DOE site, and the various storage configurations within the storage sites.

Table S–2 Sodium-Bonded Spent Nuclear Fuel Storage Locations and Configurations

Spent Nuclear Fuel Type	*Current Storage Locations and Configurations*		
	DOE Site	*Location*	*Configuration*
EBR-II driver	INEEL (ANL-W)	Radioactive Scrap and Waste Facility	Loose elements in canisters
		Hot Fuel Examination Facility	Loose elements
		Fuel Conditioning Facility	In process material [a]
EBR-II blanket	INEEL (ANL-W)	Radioactive Scrap and Waste Facility	Elements in canisters
		Fuel Conditioning Facility	In process material [a]
EBR-II driver	INEEL (INTEC)	CPP-603 basin	About 12 elements per canister
		CPP-666 basin	
Fermi-1 blanket	INEEL (INTEC)	CPP-749 underground dry well	Cut/uncut assemblies in 14 storage canisters
Fast Flux Test Facility driver	INEEL (ANL-W)	Hot Fuel Examination Facility	Loose elements
	Hanford	Fast Flux Test Facility, Buildings 405 and 403	Intact assemblies
Miscellaneous	Sandia National Laboratories/ New Mexico	Technical Area V	Experimental capsules
	SRS	Receiving Basin for Offsite Fuel	Elements
	Oak Ridge National Laboratory	Building 3525	Elements

[a] Processed as part of the EBR-II Electrometallurgical Treatment Research and Demonstration Project.

S.3 TREATMENT AND MANAGEMENT METHODS

DOE has identified several potential treatment, management, and packaging methods that could be used to prepare sodium-bonded spent nuclear fuel for disposal in a geologic repository. These are: the electrometallurgical treatment; the PUREX process; packaging in high-integrity cans; the melt and dilute process; the GMODS process; the direct plasma arc-vitreous ceramic process; and the chloride volatility process. Each of these methods is discussed below. In formulating the reasonable alternatives under the proposed action, the GMODS process, the direct plasma arc process, and the chloride volatility process were not considered sufficiently mature technologies to be included as reasonable alternatives (see Section S.6).

Direct disposal of sodium-bonded spent nuclear fuel in a geologic repository without treatment, i.e., packaging the fuel in high-integrity cans without sodium removal, has been considered in this EIS under the No Action Alternative.

S.3.1 Electrometallurgical Treatment Process

The electrometallurgical treatment process was developed at Argonne National Laboratory for processing EBR-II blanket and driver spent nuclear fuel assemblies containing metallic fuel. The process has been demonstrated for the stainless steel-clad uranium alloy fuel used in that reactor. Modifications to the process could be used for the treatment of oxide, nitride, and carbide sodium-bonded spent nuclear fuel. The electrometallurgical treatment process uses electrorefining, an industrial technology used to produce pure metals from impure metal feedstock. Electrorefining has been used to purify metal for more than 100 years.

The first step in processing sodium-bonded spent nuclear metallic fuel would be removal of the fuel elements from the fuel assemblies. The fuel elements then would be chopped into short segments and placed in stainless steel baskets in the electrorefiner, where the electrometallurgical treatment would occur. The electrorefiner would be maintained at a high temperature and would contain a molten mixture of primarily two salts. The chopped fuel elements would be lowered into the molten salt. Upon application of an electric voltage, the uranium, transuranic elements including plutonium, most of the fission products, and the sodium would dissolve into the salt. The uranium would be deposited by the current. The stainless steel cladding hulls and some of the insoluble fission products would remain in the baskets.

After a sufficient amount of spent nuclear fuel has been processed, the salt would be removed and solidified. The salt would be ground to a desired size and mixed with zeolite, a filter and ion-exchange agent, to collect certain fission products. The fission products, sodium, and transuranics, including plutonium in the salt and zeolite, would be heated so the salt becomes sorbed into the zeolite structure. Glass powder then would be added to the zeolite mixture, which would be hot-pressed to produce a ceramic high-level radioactive waste form that is expected to be suitable for disposal in a geologic repository.

The uranium would be removed and treated to remove any adhered salts. Then it would be melted (and depleted uranium would be added, if necessary), solidified to form an ingot, and further processed in a metal casting furnace to produce low-enriched uranium ingots. The stainless steel cladding hulls and the insoluble fission products would be melted in the casting furnace to produce a metallic high-level radioactive waste form that is expected to be suitable for disposal in a geologic repository.

S.3.2 PUREX Process

The PUREX process has been used extensively throughout the world since 1954 to separate and purify uranium and plutonium from fission products contained in aluminum-clad spent nuclear fuel and irradiated uranium targets. The process is not a thermal process; therefore, it takes place at low temperatures. DOE has two operating facilities at SRS, F-Canyon and H-Canyon, that use the PUREX process. Use of these facilities

for treating sodium-bonded spent nuclear fuel involves certain restrictions inherent in the design: (1) the sodium complicates the process as employed in the SRS facilities; (2) the stainless steel cladding would require significant modifications or additions to the existing facilities; and (3) the presence of alloys (e.g., zirconium) is incompatible with the SRS dissolution process. For this reason, treatment of driver sodium-bonded spent nuclear fuel is not feasible without significant modification to the existing PUREX process. However, the F-Canyon facility could be used without modifications for the blanket sodium-bonded spent nuclear fuel, if the spent nuclear fuel were declad and the sodium were removed prior to the process.

The fuel pins would be dissolved in an aqueous solution of nitric acid. The resulting solution containing uranium, plutonium, and fission products would undergo feed clarification and acidity/alkalinity adjustment. The clarified solution then would be treated via the PUREX process to produce: (1) an aqueous high-level radioactive waste containing the bulk of the fission products, americium, and neptunium; (2) a material stream containing the recovered plutonium; and (3) a material stream containing the recovered uranium. The streams would undergo a second cycle of solvent washing to further separate the residual fission products and actinides from the plutonium and uranium. The aqueous high-level radioactive waste would be processed to a borosilicate glass form. Material streams from the PUREX process would be uranium oxide, plutonium metal, and high-level radioactive waste. The uranium oxide would be stored on site as depleted uranium. The plutonium would be disposed of in accordance with the Record of Decision (75 FR 1608) for the *Surplus Plutonium Disposition Final Environmental Impact Statement* (DOE/EIS-0283).

S.3.3 High-Integrity Can Packaging

High-integrity can packaging provides substitute cladding for damaged or declad fuel, or another level of containment for intact fuel. The can is constructed of a highly corrosion-resistant material to provide corrosion protection during storage, prior to repository transfer. It also could provide long-term protection in a repository (i.e., for 1,000 or more years after repository closure with no institutional control). The high-integrity cans are placed into standardized canisters that are ready for disposal in waste packages. High-integrity cans would be used to store the sodium-bonded spent nuclear fuel on site until it can be shipped to a repository.

The analysis for packaging sodium-bonded spent nuclear fuel in high-integrity cans was performed with and without decladding and/or sodium removal. Packaging sodium-bonded blanket spent nuclear fuel in high-integrity cans with sodium removal was analyzed in the EIS under Alternative 2. Packaging sodium-bonded spent nuclear fuel in high-integrity cans without sodium removal was considered in this EIS as a direct disposal option under the No Action Alternative (see Sections S.3.8 and S.5.1). The high-integrity cans would be placed in dry storage at ANL-W. They would be placed into a standardized canister for transportation and eventual placement in waste packages in a geologic repository.

S.3.4 Melt and Dilute Process

The melt and dilute process involves chopping and melting the spent nuclear fuel and diluting it by adding depleted uranium or other metals. There are three options for the melt and dilute process that are applicable to sodium-bonded spent nuclear fuel. In the first option, bare uranium blanket spent nuclear fuel pins with the sodium removed would be melted with aluminum at SRS using technology similar to that proposed for the aluminum-clad research reactor fuel. The second and third options would be conducted at ANL-W using metallurgical technology developed for uranium and stainless steel cladding. In the second option, blanket spent nuclear fuel elements would be melted with the cladding and additional stainless steel. In the first two options, there would be no actual dilution of the fissile component of the uranium because it is present in amounts far less than in natural uranium. The third option would involve developing a new melt and dilute process capable of handling sodium volatilized from processing the chopped driver spent nuclear fuel elements

with the sodium and cladding intact. In this process option, the fuel and stainless steel would be melted under a layer of material such as molten salt.

Under the first option, declad and cleaned blanket pins would be received at SRS in aluminum canisters, each containing some 60 kilograms (132 pounds) of material. The canisters would be stored until they fit into the processing schedule. Following validation of the contents, the canisters would be loaded into a melting furnace with additional aluminum, if necessary. The furnace would operate at a very high temperature, significantly in excess of the aluminum-uranium alloy melting temperature, to initiate melting within a reasonable time frame. Volatile fission products would be captured by a series of filter banks before releasing the off-gas. A metal alloy ingot would be cast, sampled, and packaged.

Under the second option, blanket elements with the sodium removed would be loaded into a furnace crucible. A small amount of radioactive waste steel might be added to the crucible. The furnace would be heated to extremely high temperatures to melt the uranium, after which the steel would be dissolved slowly into the uranium pool. The mixture would be stirred electromagnetically to a uniform composition. Volatile fission products would be captured by a series of filter banks before releasing the off-gas. A metal alloy ingot would be cast, sampled, and packaged.

Under the third option, some of the sodium in the driver spent nuclear fuel elements would be removed in a similar manner to the sodium from blanket spent nuclear fuel elements. A melt and dilute process would be developed for driver spent nuclear fuel still containing the cladding and some metallic sodium. Chopped driver spent nuclear fuel elements would be loaded into an induction furnace and covered with a layer of low melting-temperature salt containing uranium, iron, or manganese chloride to oxidize the molten sodium. Depleted uranium would be added to reduce the enrichment. Radioactive waste steel would be added. This furnace would be operated at the same temperature as in Option 2. The sodium would react with and be captured in the flux salt, protecting the off-gas treatment filter banks. After the melt is mixed, a vacuum would be applied to complete volatilization of the salt, which would be condensed and partially reused. The salt would be stabilized in the ceramic waste form. The metal melt would be stirred and cast into an ingot, placed in a standardized canister, and stored. The process can be used for sodium-bonded spent nuclear metallic fuel. Uranium nitride, oxide, and carbide fuel types cannot be treated with this process because of their high melting points.

S.3.5 GMODS Process

The GMODS process uses oxides to convert unprocessed spent nuclear fuel directly to borosilicate glass. The basic concept is to combine unprocessed sodium-bonded spent nuclear fuel and lead-borate glass in a glass melter at a very high temperature. The uranium and plutonium in the spent nuclear fuel would be converted into oxides and dissolved in the glass. Due to the powerful dissolution and oxidation properties of the lead-borate glass melt, containment is a concern, and a water-cooled, cold-wall, induction-heated melter must be used. The process has the potential for treating both blanket and driver sodium-bonded spent nuclear fuel, if a research and development demonstration project shows that the process can deal with sodium and other factors. The waste form is borosilicate glass and would contain uranium, the transuranic elements, the fission products, and the sodium present in the sodium-bonded spent nuclear fuel.

As with all processes that dissolve or melt spent nuclear fuel, the GMODS treatment would produce radioactive off-gases. These gases would be filtered and treated.

S.3.6 Direct Plasma Arc-Vitreous Ceramic Process

In this process, the sodium-bonded spent nuclear fuel would be cut into small pieces and melted and oxidized in a rotating furnace containing molten ceramic materials at extremely high temperatures. A direct current

plasma torch would supply the energy required. Rotation would be used to keep the molten pool in the furnace. The spent nuclear fuel would be fed into the process with minimal pretreatment. Ceramic material would be added as necessary, and the mixture would be homogenized by the torch. When the spent nuclear fuel is melted and oxidized throughout the ceramic, the rotation would be slowed to allow the molten vitreous ceramic to pour out by gravity flow into canister molds. The process has the potential for treating both blanket and driver sodium-bonded spent nuclear fuel, if a research and development demonstration project shows that the process can deal with sodium and other factors.

Metallic fuel such as the EBR-II fuel would require the addition of some ceramic material. Depleted uranium could be added to reduce the uranium-235 enrichment, if necessary.

As with all processes that dissolve or melt spent nuclear fuel, the plasma arc treatment would produce radioactive off-gases. These gases would be filtered and treated.

S.3.7 Chloride Volatility Process

The chloride volatility process is an advanced treatment technology that was investigated at INEEL. The process uses the differences in the volatilities of chloride compounds to segregate major nonradiological constituents from spent nuclear fuel for the purpose of volume reduction, and isolates the fissile material to produce a glass or ceramic waste form. The major steps are: (1) extremely high-temperature chlorination and conversion of metallic fuel and cladding to gaseous chloride compounds; (2) removal of the transuranic chlorides and most of the fission products in a molten zinc chloride bed at a high temperature; (3) condensation of the other chlorides (e.g., uranium hexachloride) in a series of fluidized beds and condensers at lower temperatures; and (4) zinc chloride regeneration/recycling. The transuranics and fission product chlorides then would be converted into either fluorides or oxides for disposal.

S.3.8 Direct Disposal

For the purpose of this EIS, direct disposal of sodium-bonded spent nuclear fuel is disposal without sodium removal. The sodium-bonded spent nuclear fuel (driver and blanket) would be packaged in high-integrity containers without removing the metallic sodium. The high-integrity cans would be placed into a standardized canister designed to provide containment under repository conditions during preclosure operations. However, one of the key requirements, as specified in the current April 1999 version of DOE's Waste Acceptance Systems Requirements Document and in NRC requirements for acceptance of spent nuclear fuel or high-level radioactive waste in a geologic repository, is that it cannot contain or generate materials that are explosive, pyrophoric, or chemically reactive in a form or amount that could compromise the repository's ability to perform its waste isolation function or to satisfy its performance objective [10 CFR 60.135(b)(1)]. Under direct disposal, the pyrophoric and chemically reactive metallic sodium would not be removed and, therefore, the direct disposal option would not meet current DOE or NRC repository acceptance criteria.

S.3.9 Sodium Removal and Disposition

As discussed in Section S.2 and the preceding sections, the disposition of the metallic sodium in the sodium-bonded spent nuclear fuel varies with the treatment method. For those methods that do not require the removal of metallic sodium prior to treatment, or decladding of the fuel (e.g., the electrometallurgical process), the metallic sodium would be converted into a nonreactive salt as part of the process and would be incorporated in the high-level radioactive waste product of the process.

For the methods that require the removal of sodium prior to treatment and/or decladding of the fuel (i.e., the PUREX process, the melt and dilute process for blanket spent nuclear fuel [Options 1 and 2], and the packaging in high-integrity cans), the removed metallic sodium would be processed separately, converted into

Table S–3 summarizes sodium removal and disposition for the treatment methods addressed in this EIS.

Table S–3 Sodium Removal and Disposition by Treatment and Management Method

Treatment and Management Methods	*Decladding Required*	*Sodium Treatment*	*Sodium Disposition*
Electrometallurgical process Blanket and driver fuel	No	Stabilization	Converted into nonreactive form, as part of the process, and disposed of with the high-level radioactive ceramic waste product of the process.
High-integrity cans Blanket fuel	No	Removal	Converted into nonreactive form, separately from the process, and disposed of as low-level radioactive waste.
PUREX process Blanket fuel	Yes	Removal	Converted into nonreactive form, separately from the process, and disposed of as low-level radioactive waste.
Melt and dilute process Driver fuel	No	Removal	Converted into nonreactive form, separately from the process, and disposed of as low-level radioactive waste. The remaining sodium is separated during the process, converted to nonreactive ceramic waste form, and disposed of as high-level radioactive waste.
Blanket fuel	Yes [a]/No [b]	Removal	Converted into nonreactive form, separately from the process, and disposed of as low-level radioactive waste.
Direct disposal [c] Blanket and driver fuel	No	No	Disposed of in metallic reactive form in high-integrity cans.

[a] Melt and dilute process at SRS.
[b] Melt and dilute process at ANL-W.
[c] The direct disposal option may not meet current NRC and/or Resource Conservation and Recovery Act (RCRA) requirements.

S.4 Spent Nuclear Fuel Treatment and Management Facilities

For each alternative, DOE would require the use of existing spent nuclear fuel management facilities that provide remote handling and heavy-lifting capability, hot cells, and space to receive sodium-bonded spent nuclear fuel shipments. These facilities would prepare, treat, and/or place the sodium-bonded spent nuclear fuel in interim storage awaiting treatment as needed. Besides treating the sodium-bonded spent nuclear fuel, these facilities would provide capabilities to open the shipping containers, sample and analyze the fuel, and vacuum-dry the spent nuclear fuel. These facilities also could be used to repackage the fuel into storage canisters and place the repackaged fuel in dry interim storage to await treatment.

S.4.1 ANL-W

The ANL-W site is a center of nuclear technology development and testing. The location of ANL-W is shown in **Figure S–2**. Five nuclear test reactors have operated on the site, although the only one currently active is a small reactor used for radiography examination of experiments, waste containers, and spent nuclear fuel. Work on highly radioactive materials is conducted in the Fuel Conditioning Facility and the Hot Fuel Examination Facility, both heavily shielded hot cell facilities. Inventories of nuclear materials are maintained on site for conducting research, as well as for storage, pending decisions for further disposition.

The Fuel Conditioning Facility is one of the facilities proposed for use in treating and managing the sodium-bonded spent nuclear fuel. The Fuel Conditioning Facility was activated in 1963 and consists of two hot cells, one with an air atmosphere and the other with an inert argon gas atmosphere. Since 1990, the Fuel Conditioning Facility has undergone major reconstruction and refurbishment to meet current safety and

a nonreactive salt, and disposed of as low-level radioactive waste. Decladding and sodium removal could be done using either a mechanical process (the melt, drain, evaporate, and calcine [MEDEC] process) or a laser declad and alcohol wash process.

In the MEDEC process, blanket fuel is brought into an argon-atmosphere hot cell where the ends of the cladding are cut off. The fuel is subjected to a temperature of about 200 °C (390 °F), causing melting of the sodium, which is drained into a collection tank. The temperature is raised to about 500 °C (930 °F), which volatilizes the residual sodium. This sodium exits the fuel as sodium vapor, which is condensed in a trap and collected.

To remove the cladding after the sodium has been extracted, a special machine would be installed. This machine would mechanically push the fuel pins within the cladding out through the opening created when the cladding ends of the fuel elements previously were cut off. Experience with unirradiated blanket spent nuclear fuel at Argonne National Laboratory has shown that the pins could be mechanically pushed out of the stainless steel cladding after all the sodium has been eliminated.

For the melt and dilute process for driver spent nuclear fuel (Option 3), the sodium removed prior to the process would be treated separately, converted into a nonreactive salt, and disposed of as low-level radioactive waste. Any sodium remaining within the fuel would be removed as nonreactive salt during the melt and dilute process, stabilized in a ceramic waste form, and disposed of as high-level radioactive waste.

In the laser declad and alcohol wash process, which has been demonstrated at Rockwell International Hot Laboratory in California, a remote operation laser system and cutting machine would be used to cut the fuel elements in a predefined sequence within a hot cell. The fumes generated during the cutting process would be filtered and exhausted through an off-gas system. The fuel pins along with the cladding strips would be washed in an alcohol/water mixture to neutralize the metallic sodium and fission product (i.e., cesium) contamination. The alcohol/water solution would be partially evaporated, and the sodium/cesium alcoholates and hydroxides would be neutralized, then solidified in a grouting agent, and disposed of as low- or high-level radioactive waste, depending on the cesium content.

Several aspects of the Rockwell laser process would not meet current environmental standards and would violate the design requirements of an argon hot cell. First, the Rockwell laser process required personnel entry into the hot cell on a biweekly basis for laser maintenance and purging of the cell atmosphere to maintain a low oxygen level (less than 4 percent) and to vent alcohol/water vapors and hydrogen gas from the cell. Neither of these practices would be acceptable for argon cell operation today, in part because of stricter radiation exposure controls and a higher concentration of fission products in the remaining inventory of EBR-II blanket fuel relative to the fuel that was treated by Rockwell. Operation of an argon cell requires maintenance of a low moisture and low oxygen content atmosphere as well as limitations on liquids within the cell for criticality control. The alcohol wash process introduces a liquid which is evaporated into the cell. Second, sodium collected during previous laser decladding operations was able to be disposed of as low-level radioactive waste. The sodium collected from processing the fuel addressed by this EIS would be contaminated with cesium. If sufficient quantities of cesium were present in the sodium, this waste could not be managed as low-level radioactive waste. For the sodium to be managed as low-level radioactive waste, the sodium would have to be processed (as is done with the sodium removed from the fuel in the MEDEC process) using a currently undefined process to remove the cesium from the alcohol mixture. While criticality concerns related to high moisture content levels within a multipurpose argon cell could be eliminated by removing any stored fissile materials, frequent purging of the hot cell atmosphere and personnel entry would still be restricted by current radiation exposure controls and the high concentration of fission products involved. Only the MEDEC process was used to evaluate the various alternatives that require cleaning and/or decladding of the sodium-bonded spent nuclear fuel because of compatibility concerns about laser operation in the argon hot cell.

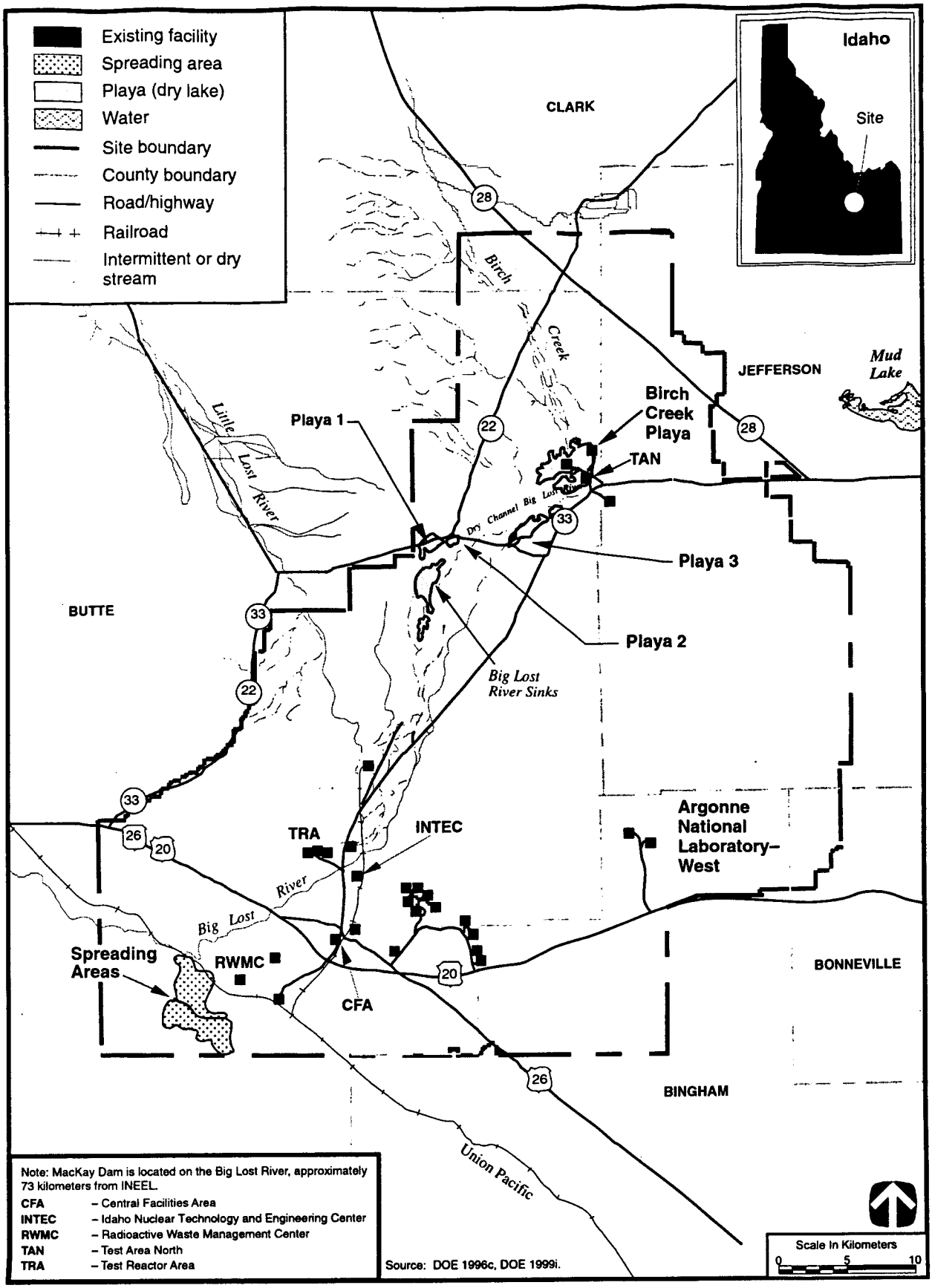
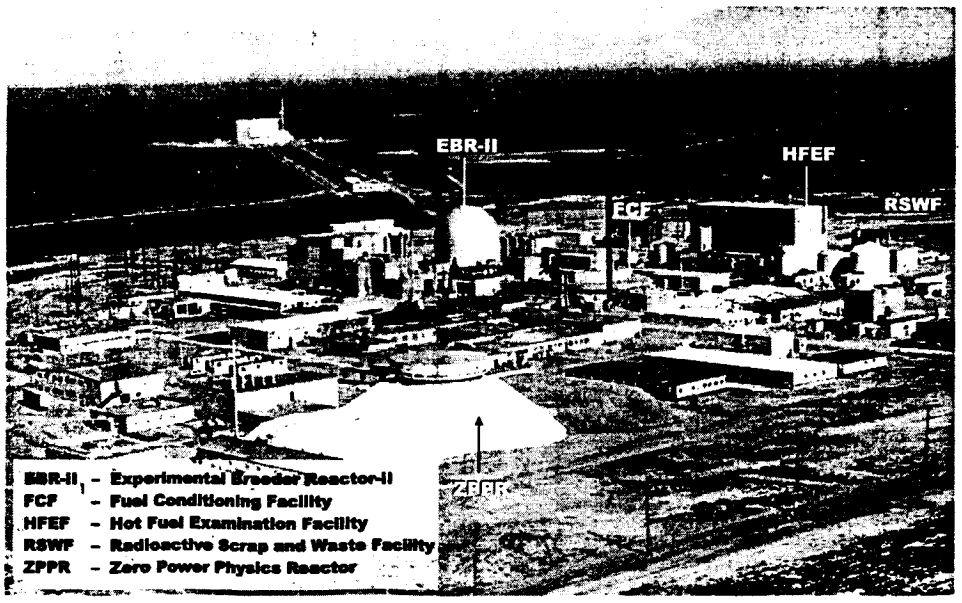


Figure S–2 Map of INEEL

environmental requirements. The hot cells enable technicians to work safely with radioactive nuclear materials from behind 1.5-meter-thick (5-foot-thick) shielding walls. The air cell is used for handling, storage, and assembly/disassembly of components. The argon cell is a much larger, doughnut-shaped hot cell where personnel can work from the outside corridor around the hot cell and work in the hot cell can be monitored from an inner shielded work space in the center of the hot cell.

The Hot Fuel Examination Facility also is proposed for use in treating and managing sodium-bonded spent nuclear fuel. The Hot Fuel Examination Facility is a hot cell complex built in the early 1970s for the preparation and examination of irradiation experiments to support a wide variety of programs and process demonstrations. A wide range of remote operations and examinations may be performed in this facility with its shielded cells, support areas, and equipment. The Hot Fuel Examination Facility is designed to be adapted to a wide variety of programs and consists primarily of two adjacent shielded cells, the main cell and the decontamination cell, in a three-story building. The decontamination cell contains an air atmosphere. The main cell contains an argon atmosphere for work involving materials such as sodium, plutonium, and other materials which could react with air. Both cells are surrounded by 1.2-meter-thick (4-foot-thick), high-density concrete to protect workers from the high radiation levels present in the hot cells. There are 21 work stations in the Hot Fuel Examination Facility, all equipped with shielded windows and remote manipulators. All in-cell equipment is carefully designed to permit remote operation and maintenance. A truck lock is located at the west end of the cell complex. The truck lock is large enough to accommodate the various trucks and fork lifts that transport the shielded casks used in the day-to-day operation of the facility. The facility recently was modified to accept truck-sized spent nuclear fuel shipping casks.



ANL-W

The Zero Power Physics Reactor Materials Storage Building at ANL-W is one of the site's primary storage facilities for uranium metal. The Zero Power Physics Reactor is currently shut down, but the facility is used for a number of projects, including a gas generation experiment. Inventories of nuclear materials stored in this facility are maintained for conducting research, as well as for storage, pending decisions for further disposition.

The Radioactive Scrap and Waste Facility at ANL-W occupies about 1.6 hectares (4 acres) and provides safe interim dry storage for spent nuclear fuel and waste generated from experiments. It is one of the facilities where the sodium-bonded spent nuclear fuel currently is stored and where high-level radioactive waste resulting from treatment of the fuel could be stored pending ultimate disposal. Located underground and 0.8 kilometers (0.5 miles) northeast of the main ANL-W facilities within the ANL-W boundary, the Radioactive Scrap and Waste Facility looks somewhat like a large parking lot on the surface. The facility has a permit issued by the State of Idaho for interim storage of mixed waste regulated under the Resource

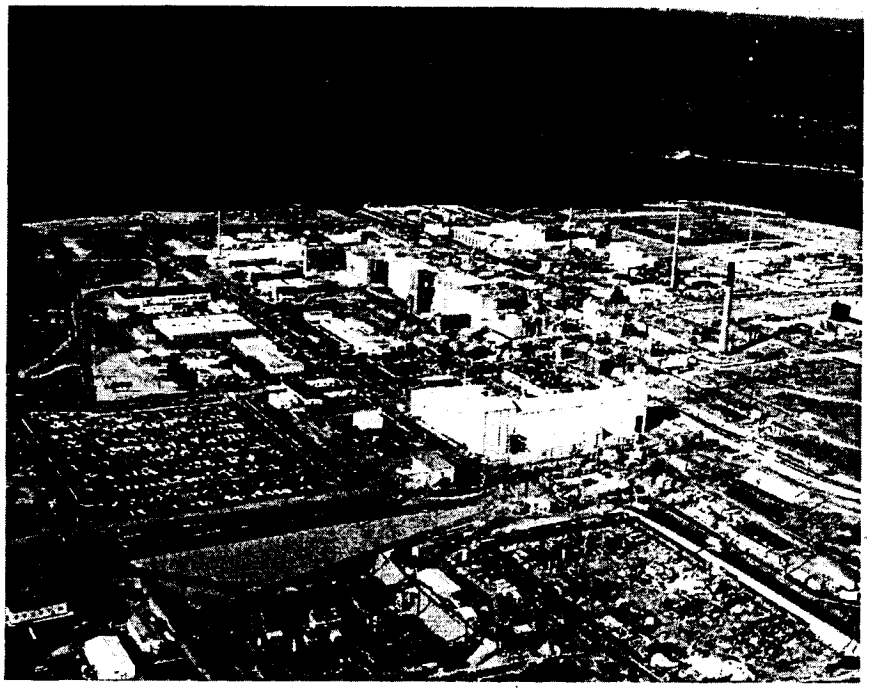
Conservation and Recovery Act (RCRA). The Radioactive Scrap and Waste Facility provides protection against corrosion for the more than 1,000 underground steel liners available for waste storage of materials handled at ANL-W.

INTEC is located northeast of the Central Facilities Area at INEEL. It is one of the sites where the sodium-bonded spent nuclear fuel currently is stored. INTEC was constructed in the 1950s to reprocess spent nuclear fuel from government reactors. In 1992, DOE announced it no longer would reprocess spent nuclear fuel. Current work at INTEC includes receiving and storing spent nuclear fuel, solidifying liquid radioactive waste, environmental restoration and decontamination and dismantling activities, and technology development. For the proposed action, the facility would be used to continue storing sodium-bonded spent nuclear fuel and for packaging the treated or untreated sodium-bonded spent nuclear fuel in standardized canisters in preparation for transport and disposal in a geologic repository. However, because it has no hot cell with an inert gas atmosphere, INTEC cannot be used for any sodium removal activities under the proposed action.

S.4.2 SRS

SRS (shown in **Figure S–3**) was constructed during the early 1950s to produce the basic materials used to fabricate nuclear weapons, primarily tritium and plutonium-239. The five reactors built on the site produced nuclear materials by irradiating target materials with neutrons. In addition, several support facilities were constructed on the site, including two chemical separation plants, a heavy water extraction plant, a nuclear fuel and target fabrication facility, and waste management facilities. As a result of changing defense requirements, all five of the original SRS production reactors have been shut down permanently. While production of new tritium will not be necessary for several years, recycling and reloading of tritium to maintain nuclear weapons reliability is a continuing site mission.

Historically, irradiated materials were moved from the SRS reactors to the two chemical separation facilities–the next step in the production process. In these facilities, known as "canyons," the irradiated fuel and target assemblies were chemically processed to separate useful products from waste. The F-Canyon at SRS could be used to chemically separate uranium from fission products in blanket spent nuclear fuel using the PUREX process. DOE uses the F-Canyon chemical separation facility and the FB-Line to stabilize spent nuclear fuel and to recycle plutonium scrap generated from facility operations and offsite sources. In September 1997, the FB-Line began a new plutonium packaging process that places stabilized plutonium in rugged, welded stainless steel cans. DOE has determined the FB-Line should be used to stabilize the plutonium recovered from spent nuclear fuel. This current program will require the FB-Line to operate until about 2002.



The F-Canyon Complex at SRS

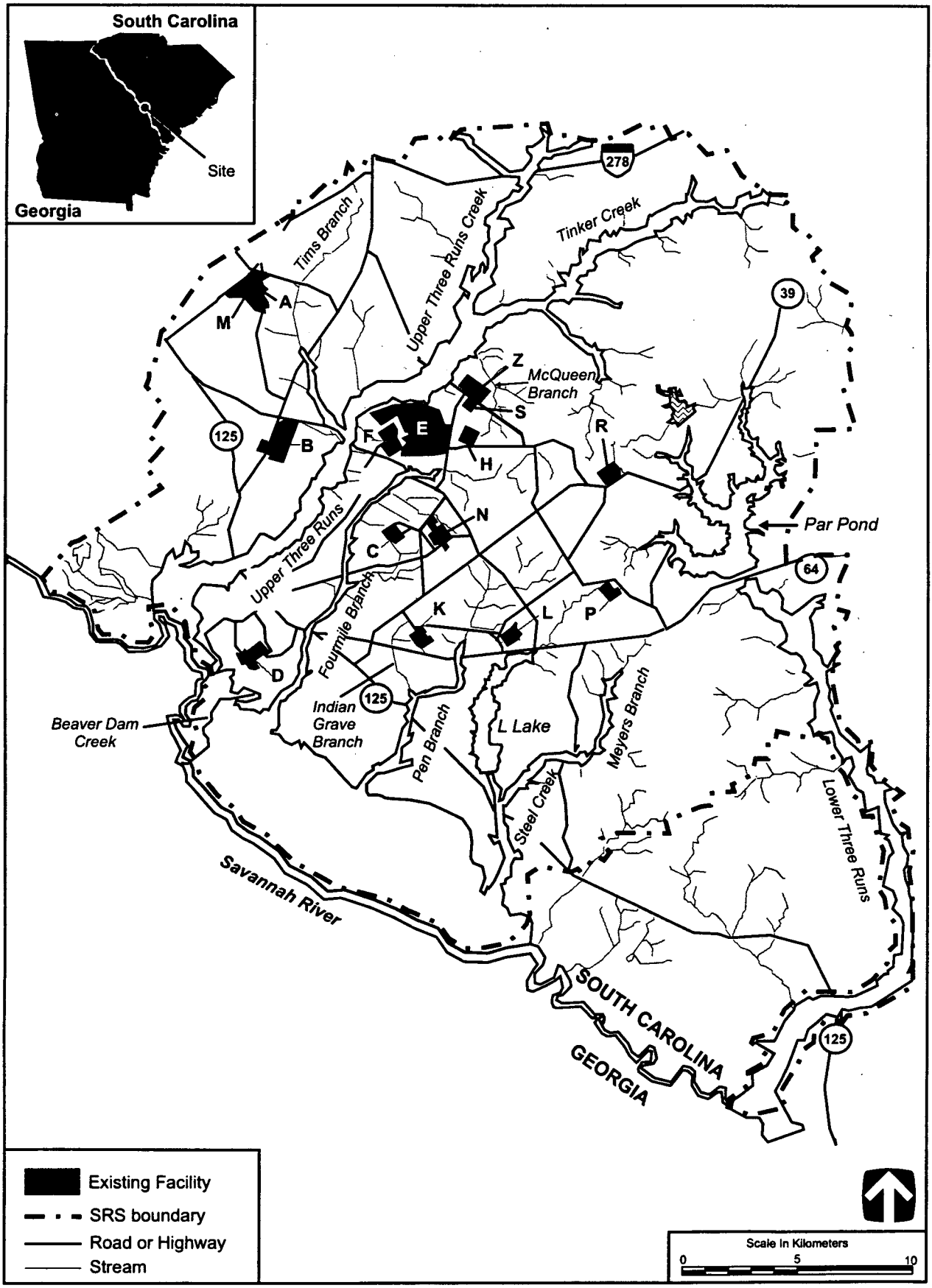


Figure S–3 Map of SRS

In the *Savannah River Site Spent Nuclear Fuel Management Draft Environmental Impact Statement*, DOE identified melt and dilute as one of the preferred methods for treating spent nuclear fuel at SRS. Building 105-L, part of the shut-down L-Reactor complex, is the SRS facility where installation of a melt and dilute process for treating spent nuclear fuel is proposed. The current mission of this facility is to store reactor components and other radioactive materials in the disassembly basin; receive and store foreign and domestic research reactor fuel in the disassembly basin; decontaminate shipping casks in the Building 105-L stack area; store contaminated moderators in tanks or drums; and compact low-level radioactive waste in a compactor.

To implement the melt and dilute technology, DOE would construct a melt and dilute facility in Building 105-L and build a dry storage facility in L-Area, near Building 105-L. DOE expects the melt and dilute option would be relatively simple to implement in Building 105-L. The major technical issue in implementing this technology would be the design of an off-gas system to capture volatilized fission products. Preliminary engineering studies indicate that the system could be designed using proven approaches for managing off-gases. The impacts from the construction of a melt and dilute facility at SRS's Building 105-L are addressed in the *Savannah River Site Spent Nuclear Fuel Management Draft Environmental Impact Statement.*

L-Reactor Complex at SRS

The Defense Waste Processing Facility, located in the S-Area, is another SRS facility that potentially could be used to treat sodium-bonded spent nuclear fuel. This facility currently is being used to convert high-level radioactive liquid waste stored at SRS into a solid borosilicate glass form that is suitable for long-term storage and disposal.

S.5 Description of Alternatives

The No Action Alternative, the alternatives under the proposed action, and the Preferred Alternative are discussed below. Although each alternative addresses both blanket and driver spent nuclear fuel, DOE considered the blanket fuel and driver fuel separately in identifying the Preferred Alternative and will do the same for the subsequent Record of Decision. In other words, DOE is considering all combinations of technologies, options, and fuel types, not only the specific combinations that are explicitly discussed in the EIS.

S.5.1 No Action Alternative

Under the No Action Alternative, all or some part of the sodium-bonded spent nuclear fuel would not be treated (no sodium would be removed), except for stabilization activities that may be necessary to prevent

Proposed Action—DOE proposes to treat and manage sodium-bonded spent nuclear fuel and facilitate its ultimate disposal in a geologic repository.

Alternative 1. *Driver and Blanket Fuel:* Under this alternative, both driver and blanket fuel would be treated using electrometallurgical treatment at ANL-W.

Alternative 2. *Driver Fuel:* Electrometallurgical treatment would be used to treat the driver fuel at ANL-W. *Blanket Fuel:* The sodium from the blanket fuel would be removed without decladding and the elements would be packaged in high-integrity cans. Sodium removal and packaging would occur at ANL-W.

Alternative 3. *Driver Fuel:* Electrometallurgical treatment would be used to treat the driver fuel at ANL-W. *Blanket Fuel:* The fuel pins would be separated from the cladding and cleaned to remove metallic sodium at ANL-W. The cleaned fuel pins would be shipped to SRS for treatment using the PUREX process at the F-Canyon facility.

Alternative 4. *Driver Fuel:* Electrometallurgical treatment would be used to treat the driver fuel at ANL-W. *Blanket Fuel:* The metallic sodium would be removed without decladding. Then the elements would be treated using the melt and dilute process. All treatment would occur at ANL-W.

Alternative 5. *Driver Fuel:* Electrometallurgical treatment would be used to treat the driver fuel at ANL-W. *Blanket Fuel:* The fuel pins would be separated from the cladding and cleaned to remove the metallic sodium at ANL-W. Then they would be shipped to Building 105-L at SRS and treated using the melt and dilute process.

Alternative 6. *Driver and Blanket Fuel:* Under this alternative, both the driver and blanket fuel would be treated at ANL-W using the melt and dilute process. The melt and dilute process would be modified slightly for each fuel type.

No Action Alternative —Under the No Action Alternative, two options were analyzed:

- Continued storage until 2035 or until development of a new or currently less mature technology to treat all or part of the sodium-bonded spent nuclear fuel
- Direct disposal of the sodium-bonded spent nuclear fuel in high-integrity cans

potential degradation of some of the spent nuclear fuel. Under the No Action Alternative, two options were analyzed: (1) the sodium-bonded spent nuclear fuel would continue to be stored until 2035 at its current location, subject only to activities dictated by the amended Record of Decision (61 FR 9441) for the Programmatic Spent Nuclear Fuel EIS and other existing site-specific NEPA documentation, or until another technology, currently dismissed as a reasonable alternative because it is less mature (e.g., GMODS or plasma arc) is developed; and (2) the sodium-bonded spent nuclear fuel would be disposed of directly in a geologic repository without treatment. The fuel would be packaged in high-integrity cans without sodium removal. Under the latter option, the sodium-bonded spent nuclear fuel at INTEC would be transported to ANL-W for packaging. As discussed in Section S.3.8, direct disposal would not meet current DOE or NRC repository acceptance criteria requirements.

A fundamental assumption made under the No Action Alternative is that the sodium-bonded spent nuclear fuel eventually will be disposed of in a manner similar to the rest of the spent nuclear fuel owned by DOE and within the time period considered over which institutional controls could reliably be assumed to be in effect. In the event that the sodium-bonded spent nuclear fuel has not been treated before 2035, the temporarily stored fuel will be removed from the State of Idaho by the year 2035. The environmental impact of the removal of untreated sodium-bonded spent nuclear fuel would be evaluated in a separate NEPA document. The continued storage of untreated sodium-bonded spent nuclear fuel in the State of Idaho or elsewhere, beyond time periods for which institutional controls could reliably be assumed to be in effect, could lead to significant impacts to

the environment and the health and safety of the public from radioactive releases caused by the gradual degradation of the fuel and its containment.

In selecting the No Action Alternative, DOE could actively pursue research and development of another treatment technology including, for example, the GMODS and plasma arc methods. These methods offer the potential to treat both blanket and driver spent nuclear fuel; they do not involve separation of uranium or plutonium; and the treatment product is expected to be suitable for disposal in a geologic repository. Reasons for not including these methods among the reasonable alternatives under the proposed action are provided in Section S.6.

S.5.2 Alternative 1: Electrometallurgically Treat Blanket and Driver Fuel at ANL-W

Under this alternative, the sodium-bonded blanket and driver spent nuclear fuel (approximately 60 metric tons of heavy metal) from ANL-W's Radioactive Scrap and Waste Facility and the Hot Fuel Examination Facility would be transported directly to the Fuel Conditioning Facility for electrometallurgical treatment. Spent nuclear fuel currently stored at INTEC would be transported to the Hot Fuel Examination Facility. This is necessary because only the Hot Fuel Examination Facility at ANL-W is capable of accepting spent nuclear fuel transportation casks. At the Hot Fuel Examination Facility, the spent nuclear fuel would be separated from the assembly hardware and packaged and transferred to the Fuel Conditioning Facility for electrometallurgical treatment. The separated hardware would be packaged and managed as low-level radioactive waste.

After treatment, the low-enriched uranium by-product would be metal-cast at the Fuel Conditioning Facility and transferred to the Zero Power Physics Reactor Materials Storage Building for storage. The remaining cladding hulls would be packaged and transferred to the Hot Fuel Examination Facility for metal casting into high-level radioactive waste and would be transferred afterward to the Radioactive Scrap and Waste Facility for storage. The electrorefiner salt containing the fission products, sodium, and transuranic elements would be transferred in metallic cans back to the Hot Fuel Examination Facility, where the ceramic waste would be produced. The ceramic waste cylinders would be packaged and transferred to the Radioactive Scrap and Waste Facility for storage. Implementing this alternative at the Fuel Conditioning Facility and the Hot Fuel Examination Facility would require the installation of some new waste handling equipment at the facilities. Electrometallurgical treatment of the sodium-bonded spent nuclear fuel at ANL-W could start as early as the year 2000, and would require approximately 12 to 13 years to process all fuel. Driver spent nuclear fuel alone would require approximately 7 years.

S.5.3 Alternative 2: Clean and Package Blanket Fuel in High-Integrity Cans and Electrometallurgically Treat Driver Fuel at ANL-W

Under this alternative, the blanket spent nuclear fuel elements (approximately 57 metric tons of heavy metal) would be packaged in high-integrity stainless steel cans at ANL-W after removal of the sodium without decladding, as discussed in Section S.3.9. Removal of the sodium from the sodium-bonded blanket spent nuclear fuel would take place at the Hot Fuel Examination Facility at ANL-W. The packaging in high-integrity cans would take place in the same facility. The high-integrity cans would be transferred to the Radioactive Scrap and Waste Facility for storage.

The sodium-bonded driver spent nuclear fuel (approximately 3 metric tons of heavy metal) would be treated using the electrometallurgical treatment process described in Section S.5.2 (Alternative 1).

Implementing this alternative at either the Fuel Conditioning Facility or the Hot Fuel Examination Facility would require the installation of equipment for sodium removal activities. In addition, some new waste handling equipment would be needed for the electrometallurgical treatment of the driver sodium-bonded spent nuclear fuel.

Packaging the blanket spent nuclear fuel in high-integrity cans could start by approximately 2003. It would take approximately six years to complete. Electrometallurgical treatment of the driver spent nuclear fuel could start in 2000 and could be completed in approximately seven years.

S.5.4 Alternative 3: Declad and Clean Blanket Fuel and Electrometallurgically Treat Driver Fuel at ANL-W; PUREX Process Blanket Fuel at SRS

Under this alternative, the blanket spent nuclear fuel pins (approximately 57 metric tons of heavy metal) would be packaged in aluminum cans and shipped to SRS for treatment using the PUREX process at the SRS F-Canyon facility. The blanket spent nuclear fuel pins would be separated from the cladding and cleaned to remove the metallic sodium at ANL-W, as discussed in Section S.3.9.

Decladding of the sodium-bonded blanket spent nuclear fuel and sodium removal could take place at the Hot Fuel Examination Facility at ANL-W. Equipment for decladding and sodium removal would need to be installed for this purpose. After decladding and sodium removal, the blanket spent nuclear fuel pins would be packaged and stored temporarily at the Hot Fuel Examination Facility to await shipment to SRS. At SRS, the cans containing blanket spent nuclear fuel pins would be unpacked at the F-Canyon facility before treatment using the PUREX process. No modifications to that facility would be needed. Waste from the process containing the fission products and transuranic isotopes other than plutonium would be transferred to the Defense Waste Processing Facility, where it would be converted to borosilicate glass logs and stored pending ultimate disposal. Approximately 260 kilograms (572 pounds) of separated plutonium would be immobilized using the "can-in-canister" technology at SRS for eventual geologic repository disposal in accordance with the Record of Decision (75 FR 1608) for the *Surplus Plutonium Disposition Environmental Impact Statement* (DOE/EIS-0283). Considering the commitment of F-Canyon to other DOE missions, PUREX processing of the blanket spent nuclear fuel would start no earlier than 2005 and would last less than one year. Decladding and sodium removal activities at ANL-W would not start earlier than 2003. Therefore, these activities would determine the length of the process.

The sodium-bonded driver spent nuclear fuel (approximately 3 metric tons of heavy metal) would be treated at ANL-W using the electrometallurgical treatment process described in Section S.5.2 for Alternative 1. As in the case of Alternative 2, electrometallurgical treatment of the driver spent nuclear fuel could start in 2000 and could be completed in approximately seven years.

S.5.5 Alternative 4: Melt and Dilute Blanket Fuel and Electrometallurgically Treat Driver Fuel at ANL-W

Under this alternative, the blanket spent nuclear fuel elements (approximately 57 metric tons of heavy metal) would be treated at ANL-W using the melt and dilute Option 2 process described in Section S.3.4. Prior to treatment, the metallic sodium would be removed without decladding at ANL-W, as discussed in Section S.3.9. Removal of the sodium from the sodium-bonded blanket spent nuclear fuel could take place at the Hot Fuel Examination Facility at ANL-W. Equipment for sodium removal would need to be installed at the facility. Equipment necessary for the melt and dilute process also would need to be installed at the facility, including the addition of a melter and an off-gas system. Metallic waste resulting from the melt and dilute process containing fission products, depleted uranium, and transuranic elements would be transferred to the Radioactive Scrap and Waste Facility for storage pending ultimate disposal. Treatment of the blanket spent nuclear fuel at ANL-W using the melt and dilute process could start as early as 2005 and could be completed in eight years.

The sodium-bonded driver spent nuclear fuel (approximately 3 metric tons of heavy metal) would be treated at ANL-W using the electrometallurgical treatment process described in Section S.5.2 for Alternative 1.

Treatment of the driver spent nuclear fuel could start as early as 2000 and could be completed in approximately seven years.

S.5.6 Alternative 5: Declad and Clean Blanket Fuel and Electrometallurgically Treat Driver Fuel at ANL-W; Melt and Dilute Blanket Fuel at SRS

Under this alternative, the blanket spent nuclear fuel pins (approximately 57 metric tons of heavy metal) would be packaged and shipped to SRS for treatment. The blanket spent nuclear fuel pins would be separated from the cladding and cleaned to remove the metallic sodium at ANL-W. The declad and cleaned blanket spent nuclear fuel pins would be received at Building 105-L at SRS and treated using the melt and dilute Option 1 process, as described in Section S.3.4. Decladding of the sodium-bonded blanket spent nuclear fuel and sodium removal would take place at the Hot Fuel Examination Facility at ANL-W. After decladding and sodium removal, the blanket spent nuclear fuel pins would be packaged and stored temporarily at the Hot Fuel Examination Facility pending shipment to SRS.

At SRS, the cans containing the blanket spent nuclear fuel pins would be unpacked at Building 105-L and the blanket spent nuclear fuel pins would be treated using the melt and dilute process. For the purpose of evaluating this alternative, it is assumed that the melt and dilute facility is operational at SRS, as proposed in the *Savannah River Site Spent Nuclear Fuel Management Final Environmental Impact Statement.* Metallic waste resulting from the melt and dilute process containing fission products, depleted uranium, and transuranic elements would be stored at the L-Area storage pending ultimate disposal.

Treatment of the blanket spent nuclear fuel at SRS would start around 2035. The facility would be operational in 2005 and already is committed to other DOE missions until 2035. If additional capacity becomes available, treatment could start as soon as 2020. The treatment process would last approximately three years. Until 2035, there would be ample time for blanket spent nuclear fuel decladding and sodium removal activities at ANL-W.

The sodium-bonded driver spent nuclear fuel (approximately 3 metric tons of heavy metal), would be treated at ANL-W using the electrometallurgical treatment process described in Section S.5.2 for Alternative 1. Treatment of the driver spent nuclear fuel at ANL-W could start in 2000 and could be completed in approximately seven years.

S.5.7 Alternative 6: Melt and Dilute Driver and Blanket Fuel at ANL-W

Under this alternative, both the sodium-bonded blanket and driver spent nuclear fuel would be treated in the Hot Fuel Examination Facility at ANL-W using Options 2 and 3 of the melt and dilute process discussed in Section S.3.4. Option 2 would be used for the blanket spent nuclear fuel and Option 3 would be used for the driver spent nuclear fuel. Uranium nitride, oxide, and carbide fuel types cannot be treated using the melt and dilute process because of their high melting points.

Removal of the sodium from the blanket spent nuclear fuel and, to the extent practical, from the driver spent nuclear fuel would take place at the Hot Fuel Examination Facility. Equipment for sodium removal activities and the melt and dilute process would need to be installed in the inert cell of the facility.

The metallic waste resulting from the melt and dilute process containing fission products, depleted uranium, and transuranic elements would be transferred to the Radioactive Scrap and Waste Facility for storage pending ultimate disposal.

The melt and dilute process at ANL-W could start as early as 2003 and would take approximately 12 years to be completed for all blanket and driver spent nuclear fuel.

S.5.8 Preferred Alternative

Council on Environmental Quality regulations (40 CFR 1502.14e) require that an agency identify its preferred alternative(s) in the final EIS. After careful consideration of public comments and programmatic, environmental, nonproliferation, and cost issues, DOE has identified electrometallurgical treatment as its Preferred Alternative for the treatment and management of all sodium-bonded spent nuclear fuel, except for the Fermi-1 blanket fuel. The No Action Alternative is preferred for the Fermi-1 blanket spent nuclear fuel. Thus, the Preferred Alternative is a combination of Alternative 1 and the No Action Alternative. This combination would result in 26 metric tons of heavy metal of EBR-II and miscellaneous spent nuclear fuel being treated using the electrometallurgical process and 34 metric tons of heavy metal of Fermi-1 blanket spent nuclear fuel remaining in storage, pending a subsequent decision on its long-term management. The environmental consequences of the Preferred Alternative are addressed in Section 4.10.

DOE will validate the cost of using alternative treatment techniques (e.g., sodium removal and placement in high-integrity cans) for the Fermi-1 blanket spent nuclear fuel. These techniques may be economically favorable for the Fermi-1 blanket spent nuclear fuel because of characteristics that distinguish it from the EBR-II spent nuclear fuel. For example, the Fermi-1 blanket spent nuclear fuel does not require the extensive safeguards and security measures that are required for the EBR-II blanket fuel. The difference in security requirements for these two types of fuel is a result of the difference in plutonium content. The EBR-II blanket fuel has 30 times more plutonium at a greater concentration than the Fermi-1 blanket fuel.

Should DOE select the Preferred Alternative in the Record of Decision, DOE would proceed with the electrometallurgical treatment of the EBR-II sodium-bonded spent nuclear fuel and monitor the results and costs while continuing the development of sodium removal techniques for the Fermi-1 blanket spent nuclear fuel. Sodium removal would increase the number of long-term management options for the Fermi-1 fuel. While EBR-II spent nuclear fuel is undergoing electrometallurgical treatment and the Fermi-1 spent nuclear fuel remains in storage, DOE has approximately four years in which to evaluate the operating experience of electrometallurgical treatment technology and further develop other alternatives for the Fermi-1 spent nuclear fuel. After these data are evaluated, DOE would decide whether to treat the Fermi-1 blanket spent nuclear fuel using electrometallurgical treatment or to use another treatment method and/or disposal technique.

Before making a decision to treat or dispose of the Fermi-1 blanket spent nuclear fuel, DOE will determine whether the analysis in this EIS is adequate to support a subsequent Record of Decision or whether additional NEPA review is required. In any case, DOE will notify the public of its preferred approach for the Fermi-1 blanket spent nuclear fuel at least 30 days before issuing a Record of Decision regarding treatment or disposal.

For several years, DOE has been actively developing electrometallurgical treatment technology specifically for the management of sodium-bonded spent nuclear fuel. Having completed a successful demonstration of electrometallurgical treatment, DOE believes that this technology has the highest probability of meeting the Department's needs for managing much of the sodium-bonded spent nuclear fuel. Electrometallurgical technology would convert the reactive fuel into ceramic and metallic waste forms, both of which are more stable than untreated sodium-bonded spent nuclear fuel. In addition, uranium would be separated from the spent nuclear fuel, blended with depleted uranium if needed to lower enrichment levels, and cast into ingots to be stored until a disposition decision is made through a separate NEPA review. Most of the plutonium would be disposed of in the ceramic waste form, with the remaining small fraction disposed of in the metallic waste form. Currently, the only waste form that has been tested and analyzed extensively under geologic repository conditions and may be accepted for repository disposal is borosilicate glass. Tests have shown the ceramic and metallic waste forms from electrometallurgical treatment may perform as well as the standard borosilicate glass waste form. The ceramic and metallic waste forms would require less storage volume than untreated spent nuclear fuel.

S.6 ALTERNATIVES CONSIDERED AND DISMISSED

In identifying the reasonable alternatives for evaluation in this EIS, two separate issues led to the determination of alternatives that were considered and dismissed: (1) the level of maturity of the alternative technologies, and (2) the level of effort required to modify an existing facility to implement a specific technology. The construction of new facilities when existing facilities are still operative was considered not a reasonable option because of impacts and cost implications. Among the treatment technologies discussed in Section S.3, the GMODS process and the direct plasma arc-vitreous ceramic process are not as mature as the electrometallurgical, melt and dilute, and PUREX processes when applied to sodium-bonded spent nuclear fuel. The GMODS and plasma arc processes both require significant and extensive research and development before they can be proven successfully to treat sodium-bonded spent nuclear fuel. The GMODS and plasma arc-vitreous ceramic processes each present specific technological challenges that cannot be answered without the construction, operation, and considerable engineering analysis of pilot-scale plants. In comparison, the melt and dilute process is being tested and evaluated, and has been selected as the Preferred Alternative for treatment of aluminum-clad spent nuclear fuel at SRS. Use of the melt and dilute process for sodium-bonded driver spent nuclear fuel only requires technology enhancement that DOE already has proposed for treating other spent nuclear fuel. In addition, unlike the other technologies that would not require new construction, both of these technologies (i.e., GMODS and plasma arc) would require the installation of large, specialized equipment in new hot cell facilities, the size and complexity of which are not determined sufficiently to allow detailed environmental impact analysis.

GMODS Process

The GMODS process, although similar to the melt and dilute process because of its thermal treatment, has not been developed beyond the laboratory scale. Several developmental steps would be required before it could be deemed a mature process. These include: detailed process development, resolution of containment concerns, testing, and a pilot plant demonstration to address technology risks (e.g., reliability and throughput).

GMODS would require large, specialized equipment to be installed in eight new large hot cell facilities. GMODS would dissolve the fuel elements or fuel assemblies entirely in a lead/lead-oxide system. An off-gas treatment system similar to that for the melt and dilute process would be required to treat the radioactive elements volatilized at about 1,000 °C (1,830 °F). The GMODS equipment could produce an intermediate waste form containing most of the actinides, fission products, and structural materials. After some preprocessing, the waste stream would be fed into the melter for the production of a new type of borosilicate glass log. These logs would contain uranium, other actinides, and structural elements, in addition to the fission products.

Because of the highly corrosive nature of the chemicals in the system, the technical feasibility of the alternative has not been established. This would add an additional degree of uncertainty to the waste estimates, as well as to the ultimate success of the fuel conditioning project.

Direct Plasma Arc-Vitreous Ceramic Process

The plasma arc-vitreous ceramic process is being used for the vitrification of low-level mixed waste. However, vitrification of spent nuclear fuel by this process is understood only on a conceptual level. The plasma arc treatment method would require large, complex equipment to be installed in a new, specially constructed hot cell facility. Such a facility could be constructed next to the Hot Fuel Examination Facility at ANL-W to secure some services. It would require the installation of equipment to cut the fuel assemblies into small pieces, a ceramic melter (furnace) to melt and oxidize the pieces at temperatures at least as high as 1,600 °C (2,900 °F), and an off-gas treatment system. As with the GMODS and melt and dilute processes, uranium and plutonium are not separated during the process. The conditioned spent nuclear fuel form would be vitreous

ceramic and would include the sodium in a stable form. As with all processes that dissolve or melt spent nuclear fuel, the plasma arc process would produce radioactive off-gases. These gases would be filtered and treated, and the filter and treatment media would be stabilized into an acceptable waste form by a yet-to-be-determined process. The process would require testing in a pilot-scale plant to address the reliability of the plasma system.

The high temperature of the process could increase the radioactive materials available for release during normal operation and accident conditions, thus increasing the exposure risk to members of the general public. Compared to other alternatives, there is substantial uncertainty about the risk from accident conditions, considering the complexity of the off-gas treatment system. Because of the high temperature, more radioactive elements would be volatilized. In addition, considerable development would be required to produce very high-temperature rotating equipment that would operate reliably in a hot cell environment.

Chloride Volatility Process

The chloride volatility process design is in an early conceptual stage. The process needs high temperatures and chlorination for volatilization and chemical reactions to separate various fission products from uranium. This treatment technology would require a very elaborate gaseous separation process with potentially significant occupational and public risks in comparison to other treatment technologies, from both the volatilized fission products and the chlorine gas.

Electrometallurgical Treatment at INEEL Test Area North

Treatment of sodium-bonded spent nuclear fuel using the electrometallurgical treatment process at INEEL's Test Area North was considered and dismissed because Test Area North would require extensive modification to treat sodium-bonded spent nuclear fuel. Implementation of this alternative would require the construction of an argon hot cell. In addition, it would require either the procurement of new equipment or the transfer of already-contaminated equipment and other systems existing at ANL-W.

Treatment of Driver or Clad Blanket Spent Nuclear Fuel Using SRS PUREX Process

As discussed in Section S.3.2, use of the PUREX process facilities at SRS for the treatment of sodium-bonded spent nuclear fuel would require the development and installation of a versatile front-end process to handle mechanical decladding, sodium removal, and zirconium sludge formation for EBR-II spent nuclear fuel. Such development does not appear justified for the sole purpose of treating the relatively small quantity of driver spent nuclear fuel.

Treatment of Driver Spent Nuclear Fuel Using SRS Melt and Dilute Process

As discussed in Section S.3.4, the treatment of driver spent nuclear fuel would require a modified melt and dilute process that would be capable of handling the sodium volatilized from processing chopped driver spent nuclear fuel elements with the cladding intact. To accomplish this at SRS, significant design changes would be required from the process that DOE has proposed for the aluminum-clad spent nuclear fuel, which does not contain sodium. These design changes do not appear justified for the sole purpose of treating the relatively small quantity of driver spent nuclear fuel.

Treatment Using INEEL PUREX Process

Sodium-bonded spent nuclear fuel from EBR-II was being processed at the Idaho CPP (now INTEC) using a PUREX process. DOE stopped processing at INTEC as a matter of policy in 1992, and the facility was permanently shut down. Reactivation of the facility is not practical and the alternative was dismissed.

S.7 AFFECTED ENVIRONMENT

INEEL is located on approximately 230,700 hectares (570,000 acres) in southeastern Idaho and is 55 kilometers (34 miles) west of Idaho Falls, 61 kilometers (38 miles) northwest of Blackfoot, and 35 kilometers (22 miles) east of Arco. It is located primarily within Butte County, but portions of the site are also in Bingham, Jefferson, Bonneville, and Clark counties. Much of INEEL is open space that has not been designated for specific use. Land use at INEEL includes facility operations, grazing, general open space, and infrastructure (such as roads). The site lies in a cool desert ecosystem dominated by shrub-steppe vegetative communities. Developed portions of INEEL occur within the 93,000-hectare (230,000-acre) central core area of the site. ANL-W is located in the southeast portion of the central core area, about 7 kilometers (4.3 miles) northwest of the nearest site boundary, and is designated as a testing center for advanced technologies associated with nuclear power systems. Other than internal modification to existing facilities, no new construction would take place within ANL-W for any of the proposed alternatives.

SRS is located on about 80,130 hectares (198,000 acres) in southwest South Carolina. The site is 40 kilometers (25 miles) southeast of Augusta, Georgia, and 19 kilometers (12 miles) south of Aiken, South Carolina. It is bordered by the Savannah River to the southwest and includes portions of three South Carolina counties: Aiken, Allendale, and Barnwell. Land use at SRS includes forest and undeveloped areas, water and wetlands, and developed facilities. Land use in F-Area is classified as heavy industrial, with facilities that historically have been associated with chemical and physical processes used to separate uranium, plutonium, and fission products. Land use in L-Area also is classified as heavy industrial, with facilities that historically have been associated with nuclear materials production for national defense. Other than internal modification to existing facilities, no new construction would take place within SRS for any of the proposed alternatives.

S.8 CONSEQUENCES

This section summarizes the environmental impacts associated with the No Action Alternative and the six alternatives under the proposed action that are evaluated in detail in this EIS. For the No Action Alternative and the six alternatives evaluated, the proposed facilities already exist. Except for internal building modifications and new equipment installation, no construction activities would be required. Therefore, DOE has determined the proposed action would have minimal or no impacts on land resources, visual resources, noise, geology and soils, ecological resources, and cultural and paleontological resources. These areas were not evaluated in detail in the EIS. The information presented below is based on Chapter 4 of the EIS, which provides a detailed discussion of the impacts on the potentially affected environmental areas. Such environmental areas include: air quality, water resources, socioeconomics, public and occupational health and safety, environmental justice, waste management, and transportation. A summary of the environmental impacts for the No Action Alternative and the six alternatives under the proposed action is presented as **Table S–4**.

As explained below and shown in Table S–4, for the alternatives evaluated the analyses showed that there would be no significant impacts on air quality, water resources, socioeconomics, public and occupational health and safety, environmental justice, and transportation. The radiological and nonradiological gas and liquid releases, as well as the associated exposures to workers and the public, would be well below regulatory standards and guidelines and, therefore, no mitigation measures would be warranted. Finally, the environmental impact analysis indicates that there are no significant impacts that would discriminate one alternative over another from an environmental impact point of view.

Radiological Health Effects Risk Factors Used in this EIS

Health impacts of radiation exposure, whether from sources external or internal to the body, are generally identified as "somatic" (i.e., affecting the exposed individual) or "genetic" (i.e., affecting descendants of the exposed individual). Radiation is more likely to produce somatic effects (i.e., induced cancers) than genetic effects. Except for leukemia, which can have an induction period (time between exposure to carcinogen and cancer diagnosis) of as little as 2 to 7 years, most cancers have an induction period of more than 20 years. Because of the delayed effect, the cancers are referred to as "latent" cancers.

For a uniform irradiation of the body, the incidence of cancer varies among organs and tissues; the thyroid gland and skin demonstrate a greater sensitivity than other organs. Such cancers, however, also produce comparatively low mortality rates because they are relatively amenable to medical treatment. Because fatal cancer is the most probable serious effect of environmental and occupational radiation exposure, estimates of cancer fatalities, rather than cancer incidents, are presented in this EIS.

The number of latent cancer fatalities is estimated using risk factors determined by the International Commission on Radiological Protection. A risk factor is the probability that an individual would incur a latent cancer fatality during his or her lifetime if the individual receives a unit of radiation dose (1 rem). The risk factor for workers is 0.0004 (latent cancer fatality per rem), and 0.0005 (latent cancer fatality per rem) for individuals among the general public. The risk factor for the public is slightly higher because the public includes infants and children, who are more sensitive to radiation than adults.

Examples:

The latent cancer fatality risk for an individual (nonworker) receiving a dose of 0.1 rem would be 0.00005 (0.1 rem x 0.0005 latent cancer fatality per rem). This risk can also be expressed as "0.005 percent chance" or "one chance in 20,000."

The same concept is used to calculate the latent cancer fatality risk from exposing a group of individuals to radiation. The latent cancer fatality risk for individuals in a group of 100,000, each receiving a dose of 0.1 rem, would be 0.00005, as indicated above. This individual risk, multiplied by the number of individuals in the group, expresses the number of latent cancer fatalities that could occur among the individuals in the group. In this example, the number would be 5 latent cancer fatalities (100,000 x 0.00005). A number of latent cancer fatalities less than 1 means that the radiation exposure is not sufficient to cause a single latent cancer fatality among the members of the group. In this case, the risk is expressed as a probability that a single latent cancer fatality would occur among the members of the group. For example, 0.05 latent cancer fatalities can be stated as " there is one chance in 20 (1/0.05) that one latent cancer fatality would occur among the members of the group."

The EIS provides estimates of probability of a latent cancer fatality occurring for the involved and noninvolved workers, the maximally exposed offsite individual, an average individual, and the general population. These categories are defined as follows:

Involved worker—An individual worker participating in the operation of the facilities.

Noninvolved worker—An individual worker at the site other than the involved worker.

Maximally exposed offsite individual—A member of the public residing at the site boundary who could receive the maximum dose of radiation or exposure to hazardous chemicals.

Average individual—A member of the public receiving an average dose of radiation or exposure to hazardous chemicals.

Population—Members of the public residing within an 80-kilometer (50-mile) radius of the facility.

Air Quality

The proposed action and either option of the No Action Alternative would have a negligible impact on existing air quality at ANL-W, INTEC, or SRS. Radiological emissions also would be low and well below regulatory concern (see the public and occupational health and safety discussion, below).

Water Resources

Surface water is not used at ANL-W or INTEC, and this would not change under the proposed action or either option of the No Action Alternative. Groundwater use, primarily domestic consumption, would remain at current levels under the proposed action and could decrease with a reduction in workers at ANL-W under the No Action Alternative.

No changes are expected in liquid effluent discharges under the proposed action or the No Action Alternative at ANL-W. There currently are no discharges to surface waters (radiological or nonradiological), except for discharges of nonhazardous liquid waste, which are monitored and subject to Idaho Land Permit Application requirements.

Potential radioactive liquid effluent has been identified for the PUREX process at SRS under Alternative 3. Table S–4 indicates some small quantities of tritium and other radionuclides (including strontium, cesium, promethium, and isotopes of uranium and plutonium) would be released. No radioactive liquid effluent has been identified for the melt and dilute process at SRS under Alternative 5.

Socioeconomics

Under either option of the No Action Alternative, there could be a reduction of approximately 350 workers at ANL-W. This reduction could result in the loss of 940 additional indirect jobs in the economic region. The reduction would take place over time; therefore, the No Action Alternative would not result in any noticeable changes in the existing regional economy, housing characteristics, or community services.

All the alternatives under the proposed action assume that the treatment and management of the sodium-bonded spent nuclear fuel at ANL-W or SRS would not require an additional work force, but the activities would keep the work force from being reduced. Therefore, there would be no changes to the socioeconomic conditions in the vicinity of either ANL-W or SRS.

Public and Occupational Health and Safety

The risk to the health and safety of the workers and the public under the proposed action or either option of the No Action Alternative would be from the potential exposure to radiological or hazardous chemical emissions during normal operation or accident conditions. As indicated below and in Table S-4, this risk would be very small. The health effects presented below are given in terms of total project effects to account for the variations in annual exposures within an alternative as well as variations in project durations between alternatives.

Radiological Exposures

The maximum project total impact to the population within 80 kilometers (50 miles) from radioactive releases under the proposed action would be 0.000012 (Alternative 6) additional latent cancer fatalities, or one chance in 83,000 for one additional latent cancer fatality within the exposed population, compared to 6.5×10^{-6} latent cancer fatalities for the No Action Alternative. The project total dose corresponding to this impact would be

0.024 person-rem. For comparison purposes, the collective dose to this population from natural background radiation for a single year would be 86,250 person-rem.

The maximum project total impact to the maximally exposed offsite individual under the proposed action would be 2.0×10^{-9} (Alternative 6), or one chance in 500 million that this individual would develop a fatal cancer, compared to 1.1×10^{-9} (one chance in 900 million) for the No Action Alternative. The maximum project total dose to this individual under the proposed action would be 0.004 millirem. The regulatory limit for offsite individuals is 10 millirem per year from air exposure and 100 millirem per year from all pathways.

The maximum project total impact to the workers under the proposed action at ANL-W or SRS would be 0.13 additional latent cancer fatalities, or one chance in 7 that the work force would experience an additional latent cancer fatality from this exposure, compared to 0.084 (one chance in 12) additional latent cancer fatalities expected from the No Action Alternative. The maximum project total dose under the proposed action would be 319 person-rem for the worker population. DOE's administrative limit for individual worker exposure is 2,000 millirem per year.

Hazardous Chemical Exposures

Hazardous chemical impacts resulting from the proposed action or either option under the No Action Alternative under normal conditions would be small because any routine emissions of hazardous chemicals would be very low.

Hazardous chemical impacts under accident conditions, evaluated in terms of Emergency Response Planning Guideline values, indicate that under the worst postulated accident conditions, no adverse health effects to a worker or the maximally exposed offsite individual at either ANL-W or SRS would occur.

Environmental Justice

As discussed above, the impacts from the proposed action or either option under the No Action Alternative on the health and safety of the public would be very small, regardless of the racial and ethnic composition of the population and independent of the economic status of the individuals comprising the population in the region.

Waste Management

For both options under the No Action Alternative, various types of waste would continue to be generated at ANL-W, including low-level radioactive, transuranic, mixed, hazardous, and nonhazardous waste. These waste types are associated with the operation of the facilities where the sodium-bonded spent nuclear fuel is stored. High-level radioactive waste in metallic and ceramic forms generated as a result of completing the Electrometallurgical Treatment Research and Demonstration Project waste processing would be stored at the Radioactive Scrap and Waste Facility pending disposal. Finally, some additional low-level radioactive waste and transuranic waste would be generated from the deactivation of the demonstration project. The volumes of these waste types are presented in Table S–4.

Table S–4 also presents a comparison of the volumes of high-level radioactive, low-level radioactive, and transuranic waste generated by each of the alternatives under the proposed action. The volume of high-level radioactive waste to be disposed of in the repository under the proposed action would be reduced by 37 to 84 percent compared to the volume of spent nuclear fuel under the direct disposal option of the No Action Alternative. On the other hand, with the exception of Alternative 2, all alternatives under the proposed action would generate higher volumes of low-level radioactive and transuranic waste than the No Action Alternative.

All of the alternatives under the proposed action would either remove or convert the metallic sodium into a nonreactive form.

With respect to disposability and waste acceptance criteria, only the borosilicate glass waste form generated by Alternative 3 for blanket spent nuclear fuel has been tested and analyzed extensively under conditions relevant to a geologic repository. It is expected that other waste forms (e.g., ceramic, metallic) and possibly high-integrity cans not containing metallic sodium would be suitable for repository disposal.

Transportation

Incident-free transportation activities under the proposed action for alternatives not involving transportation of fuel to SRS would result in 0.000011 latent cancer fatalities among transportation workers (Alternative 6) and 0.0001 latent cancer fatalities in the total affected population (Alternative 6), compared to 1.2×10^{-6} latent cancer fatalities for transportation workers and 0.000011 latent cancer fatalities for the affected population under the No Action Alternative.

The latent cancer fatality risk to the affected population from postulated accidents under the proposed action and the No Action Alternative would be less than 1.0×10^{-10}, or one chance in 10 billion.

Incident-free transportation activities under the proposed action (for alternatives involving transportation of fuel to SRS) would result in 4.7×10^{-7} additional latent cancer fatalities among transportation workers and 6.0×10^{-6} additional latent cancer fatalities among the affected population over the duration of the transportation activities. These are risks from radiological causes. There would be approximately 0.00039 additional latent fatalities related to nonradiological (traffic) causes among affected urban populations along the transportation route.

The latent cancer fatality risk to the affected population from postulated transportation accidents would be 1.7×10^{-9}, or one chance in 588 million. The nonradiological (traffic) fatalities would be 0.0018.

S.9 Cumulative Impacts

The Council on Environmental Quality regulations implementing NEPA procedural provisions define cumulative impacts as the impacts on the environment which result from the incremental impact of the action when added to other past, present, and reasonably foreseeable future actions, regardless of what agency (Federal or nonfederal) or person undertakes such other actions (40 CFR 1508.7). The cumulative impacts analysis is based on the incremental contribution from the maximum impacts from the proposed action added to baseline conditions at ANL-W and SRS and the maximum impacts from other onsite and offsite past, present, and other reasonably foreseeable future actions.

As indicated in Section S.8, the proposed action would have minimal or no impacts on land, visual, noise, geology and soils, ecological, and cultural and paleontological resources. The contributory effect from the proposed action on these resources would be so small that their potential contribution to cumulative impacts at ANL-W, INEEL, and SRS would be negligible.

Small but finite impacts have been identified for air and water resources and public and worker health and safety at ANL-W, INEEL, and SRS as a result of potential releases of radiological effluent and waste management activities. Treatment would transform the sodium-bonded spent nuclear fuel into another disposable waste form and small amounts of transuranic and low-level radioactive waste would be generated in the process.

The cumulative impact analysis in the EIS indicates that incremental impacts to public and worker health and safety from the proposed action, when added to the impacts from other past, present, and reasonably foreseeable future actions at ANL-W, INEEL, and SRS, would result in cumulative impacts well below applicable regulatory limits and guidelines. In addition, the projected amount of high-level radioactive, low-level radioactive, and transuranic waste generation, when added to the amount of waste generation from other past, present, and reasonably foreseeable future actions at these sites, would not require any additional treatment or storage capacities beyond the current and planned capacities at these sites.

Current and reasonably foreseeable future actions at ANL-W and INEEL include the Advanced Mixed Waste Treatment Program and the Idaho High-Level Radioactive Waste and Facilities Disposition Program, as described in the corresponding EISs. In addition, INEEL is also being considered as a possible site for the production of plutonium-238. This action and its cumulative effects will be addressed at a later date.

Reasonably foreseeable future actions at SRS include the Surplus Plutonium Disposition Program at SRS, the Interim Management of Nuclear Materials, the construction and operation of a Tritium Extraction Facility, the Surplus Highly Enriched Uranium Disposition Program, the management of plutonium and scrub alloy residues, and other DOE Complex miscellaneous components as described in the SRS Spent Nuclear Fuel Management EIS.

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Table S–4 Summary of Environmental Consequences for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel

Resource/Material Categories	*No Action*	*Alternative 1*	*Alternative 2*
	Two Options Were Considered: a. Continued Storage Until 2035, b. Direct Disposal Without Sodium Removal	**Electrometallurgically Treat Blanket and Driver Fuel at ANL-W**	**Clean [a] and Package Blanket Fuel in High-Integrity Cans and Electrometallurgically Treat Driver Fuel at ANL-W**
Air Quality - Radiological air emissions (curies per year)	Negligible impact 811 [c]	Negligible impact Tritium: 770 Krypton-85: 11,600	Negligible impact Tritium: 809 Krypton-85: 11,860
Water Resources - Radiological liquid effluent (curies per year)	No impact No liquid effluent	No impact No liquid effluent	No impact No liquid effluent
Socioeconomics	Loss of 350 direct jobs and 940 indirect jobs; no noticeable impact	Work force maintained; No impact	Work force maintained; No impact
Public and Occupational Health and Safety Risk			
• Project duration	35 years	13 years	9 years
• Normal operations (annual) [d]	**LCF [e]**	**LCF [e]**	**LCF [e]**
- Population	7.5×10^{-7}	1.4×10^{-6}	1.5×10^{-6}
- MEI	1.3×10^{-10}	1.7×10^{-10}	1.9×10^{-10}
- Average individual	3.1×10^{-12}	5.8×10^{-12}	6.2×10^{-12}
- Worker population	0.0088	0.0088	0.0088
- Average worker	0.000024	0.000024	0.000024
• Normal operations (project total) [d]			
- Population	6.5×10^{-6}	8.2×10^{-6}	8.3×10^{-6}
- MEI	1.1×10^{-9}	1.0×10^{-9}	1.0×10^{-9}
- Worker population	0.084	0.13	0.092
• Hazardous chemicals			
- MEI	None	None	None
Environmental Justice	No disproportionately high and adverse impact to minority or low-income populations		
Waste Management (cubic meters)			
• High-level radioactive waste	152 (Direct disposal volume) [f]	81.1	43.2 [g]
• Low-level radioactive waste	904	861	733.7
• Transuranic waste	12	14.1	10.7
Transportation Risk			
• Incident-free	**LCF [e]**	**LCF [e]**	**LCF [e]**
- Population	0.000011	0.000016	0.000016
- Workers	1.2×10^{-6}	1.8×10^{-6}	1.7×10^{-6}

LCF = Latent Cancer Fatalities; MEI = Maximally Exposed Offsite Individual.

[a] Clean means sodium removal.

[b] Over a period of six months.

[c] Represents total curies for 35 years; tritium: 51 curies; krypton-85: 760 curies; iodine-129: 0.000018 curies.

[d] Annual value represents the maximum impact in a single year. Project total value represents the total impact to account for the variations in annual impact over the project duration.

Alternative 3		*Alternative 4*	*Alternative 5*		*Alternative 6*
Clean [a], Declad, and Package Blanket Fuel and Electrometallurgically Treat Driver Fuel at ANL-W	**PUREX Process Blanket Fuel at SRS [b]**	**Clean [a] and Melt and Dilute Blanket Fuel and Electrometallurgically Treat Driver Fuel at ANL-W**	**Clean [a], Declad, and Package Blanket Fuel and Electrometallurgically Treat Driver Fuel at ANL-W**	**Melt and Dilute Cleaned and Declad Blanket Fuel at SRS**	**Melt and Dilute Blanket and Driver Fuel at ANL-W**
Negligible impact Tritium: 809 Krypton-85: 11,860	Negligible impact Tritium: 162 Krypton-85: 1,187	Negligible impact Tritium: 809 Krypton-85: 11,860	Negligible impact Tritium: 809 Krypton-85: 11,860	Negligible impact Tritium: 54 Krypton-85: 396	Negligible impact Tritium: 2,162 Krypton-85: 32,650
No impact No liquid effluent	Negligible impact Tritium: 1.54 Other: less than 0.022	No impact No liquid effluent	No impact No liquid effluent	No impact No liquid effluent	No impact No liquid effluent
Work force maintained; no impact	Work force maintained; no impact	Work force maintained; no impact	Work force maintained; no impact	Work force maintained; no impact	Work force maintained; no impact
9 years	Less than 1 year	13 years	9 years	3 years	12 years
LCF [c]	**LCF [c]**	**LCF [c]**	**LCF [c]**	**LCF [c]**	**LCF [c]**
1.5×10^{-6}	0.000010	1.5×10^{-6}	1.5×10^{-6}	3.8×10^{-6}	6.1×10^{-6}
1.9×10^{-10}	2.6×10^{-10}	1.9×10^{-10}	1.9×10^{-10}	5.0×10^{-11}	1.0×10^{-9}
6.2×10^{-12}	1.2×10^{-11}	6.2×10^{-12}	6.2×10^{-12}	5.5×10^{-12}	2.6×10^{-11}
0.0088	0.015	0.0088	0.0088	0.02	0.0088
0.000024	0.0001	0.000024	0.000024	0.0002	0.000024
8.3×10^{-6}	0.000010	8.3×10^{-6}	8.3×10^{-6}	0.000011	0.000012
1.0×10^{-9}	2.6×10^{-10}	1.0×10^{-9}	1.0×10^{-9}	1.5×10^{-10}	2.0×10^{-9}
0.092	0.015	0.13	0.092	0.06	0.12
None	None	None	None	None	None
No disproportionately high and adverse impact to minority or low-income populations					
23.6 (18 at ANL-W; 5.6 at SRS)		63.6	94.62 (18 at ANL-W; 76.62 at SRS)		86
2,960.5 (770.5 at ANL-W; 2,190 at SRS)		845	1,178.5 (770.5 at ANL-W; 408 at SRS)		924
100.7 (10.7 at ANL-W; 90 at SRS)		12.8	27.2 (10.7 at ANL-W; 16.5 at SRS)		14.1
LCF [c]	**LCF [c]**	**LCF [c]**	**LCF [c]**	**LCF [c]**	**LCF [c]**
0.000015	6.0×10^{-6}	0.000072	0.000015	6.0×10^{-6}	0.0001
1.6×10^{-6}	4.7×10^{-7}	7.9×10^{-6}	1.6×10^{-6}	4.7×10^{-7}	0.000011

[c] Dose to the MEI, average individual, and the population (in person-rem) can be found by dividing the corresponding LCF values by 0.0005. Dose to the average worker and worker population (in person-rem) can be found by dividing the corresponding LCF values by 0.0004. The regulatory dose limit for offsite individuals (public) is 0.010 rem per person per year from air exposures, and 0.1 rem per person per year for all pathways. The administrative control limit for an individual worker at a DOE site is 2 rem per person per year.

[f] Includes 142 cubic meters of spent nuclear fuel.

[g] Includes 25.2 cubic meters of spent nuclear fuel.

S.10 GLOSSARY

Background Radiation — Ionizing radiation present in the environment from cosmic rays and natural sources in the Earth; background radiation varies considerably with location. The U.S. average background radiation is 300 millirem per year.

Blanket Fuel — Those fuel tubes or elements composed of depleted or natural enrichment of uranium, placed at the perimeter of the reactor core, and used to breed the fissile material plutonium-239 or used as shielding.

Borosilicate Glass — Glass typically containing approximately 20 to 40 weight percent waste oxides, 40 to 65 weight percent silica, 5 to 10 weight percent boron oxide, and 10 to 20 weight percent alkali oxides, plus other oxide constituents.

Breeder Reactor — A type of nuclear reactor that creates more fissile fuel than it uses.

Burnup — A term used to indicate the amount of fuel consumed during the irradiation process. The percentage of heavy metal atoms fissioned or the thermal energy produced per mass of fuel (usually measured in megawatt days per ton (MWd/t).

Canister — The structure surrounding the waste form (e.g., high-level radioactive waste immobilized in borosilicate glass) that facilitates handling, storage, transportation, and/or disposal. A canister is a metal receptacle that (1) acts as a pour mold for solidified high-level radioactive waste, and (2) for spent nuclear fuel, may provide structural support for intact spent nuclear fuel, loose rods, nonfuel components, or confinement of radionuclides.

Cladding — The outer jacket of fuel elements, usually made of aluminum, stainless steel, or zirconium alloy, used to prevent fuel corrosion and retain fission products during reactor operation or to prevent releases into the environment during storage.

Conditioning — Any process which prepares or treats spent nuclear fuel or high-level radioactive waste for storage, transportation, or disposal in accordance with regulatory requirements (e.g., melt and dilute product).

Curie — unit of radioactivity equal to 37 billion disintegrations per second; also a quantity of any nuclide or mixture of nuclides having 1 curie of radioactivity.

Decladding — The process of mechanically removing the cladding from the fuel pin in a fuel element.

Depleted Uranium — Uranium with a smaller percentage of uranium-235 than the 0.7 percent found in natural uranium. It is a by-product of the uranium enrichment process, during which uranium-235 is collected from one batch of uranium, thereby depleting it, and added to another batch to increase its concentration of uranium-235.

Dilute — To reduce the concentration of a substance by adding another material to it.

Disposal — The isolation of radioactive waste from the accessible environment, as defined in 10 CFR 60.2. Disposal means the emplacement in a repository of high-level radioactive waste, spent nuclear fuel, or other highly radioactive material with no foreseeable intent of recovery, whether or not such emplacement permits the recovery of such waste.

Disassembly — Removal of the fuel elements from the fuel assembly.

Dose — The energy imparted to matter by ionizing radiation. The unit of absorbed dose is the rad.

Driver Fuel — Those fuel tubes or elements composed of enriched uranium, placed at the center of the reactor core, and used to sustain the fission chain reaction.

Effluent (liquid) — Wastewater, treated or untreated, that flows out of a treatment plant, sewer, or industrial outfall; generally refers to waste discharged into surface waters.

Emission — A material discharged into the atmosphere from a source operation or activity.

Fission Products — Elements (primary fission products) formed by the fission of heavy elements; also, the elements formed by the decay of the primary fission products, many of which are radioactive.

Fissium — An alloy of molybdenum, ruthenium, rhodium, palladium, zirconium, and niobium.

Fuel Assembly — A cluster of fuel elements (or rods). Approximately 200 fuel assemblies make up a reactor core.

Fuel Element — Nuclear reactor component that includes the sealed fissile material (fuel pin) and the cladding.

Fuel Pin —The uranium metal or alloy that undergoes fission in a nuclear reactor (without cladding).

Geologic Repository — A system that is intended to be used for, or may be used for, the disposal of radioactive waste and spent nuclear fuel in excavated geologic media. A geologic repository includes (1) the geologic repository operations area, and (2) the portion of the geologic setting that provides isolation. A near-surface disposal area is not a geologic repository.

Hazardous Waste — Any solid waste (also can be semisolid or liquid, or contain gaseous material) having the characteristics of ignitability, corrosivity, toxicity, or reactivity, defined by the Resource Conservation and Recovery Act and identified or listed in 40 CFR 261 or by the Toxic Substances Control Act.

High-Level Radioactive Waste — The (1) highly radioactive waste material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid waste derived from such liquid waste that contains fission products in sufficient concentrations; and (2) other highly radioactive material that is determined, consistent with existing law, to require permanent isolation.

Latent Cancer Fatalities — Fatalities associated with acute or chronic environmental exposure to chemicals or radiation that occur from delayed effects years after exposure.

Low-Level Radioactive Waste — Waste that contains radioactivity but is not classified as high-level radioactive waste, transuranic waste, spent nuclear fuel, or by-product material as defined by Section 11e (2) of the Atomic Energy Act of 1954, as amended.

Management — As used in this EIS, the stabilization and interim storage of sodium-bonded spent nuclear fuel pending final disposition.

Maximally Exposed Offsite Individual — Hypothetical individual defined to allow dose or dosage comparison with numerical criteria for the public. This individual is located at the point on the DOE site boundary nearest to the facility in question. A hypothetical person who potentially could receive the maximum dose of radiation or hazardous chemicals.

Metric Tons of Heavy Metal — Quantities of unirradiated and spent nuclear fuel 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.

Millirem — One thousandth of a rem.

Mixed Waste — Waste that contains both "hazardous waste" and "radioactive waste" as defined in this glossary.

Normal Operation— All activities associated with a facility mission, whether operation, maintenance, storage, etc., which are carried out within a defined envelope. This envelope can be design process conditions, performance in accordance with procedures, and so forth.

Packaging — With regard to hazardous or radionuclide materials, the assembly of components necessary to ensure compliance with Federal regulations for transportation. It may consist of one or more receptacles, absorbent materials, spacing structures, thermal insulation, radiation shielding, and devices for cooling or absorbing mechanical shocks. The vehicle tie-down system and auxiliary equipment may be designated as part of the packaging.

Person-Rem — The unit of collective radiation dose to a given population; the sum of the individual doses received by a population segment.

Pyrophoric — Highly susceptible to spontaneous ignition and sustained combustion.

Radioactive Waste — Materials from nuclear operations that are radioactive or are contaminated with radioactive materials, and for which use, reuse, and/or recovery would be impractical.

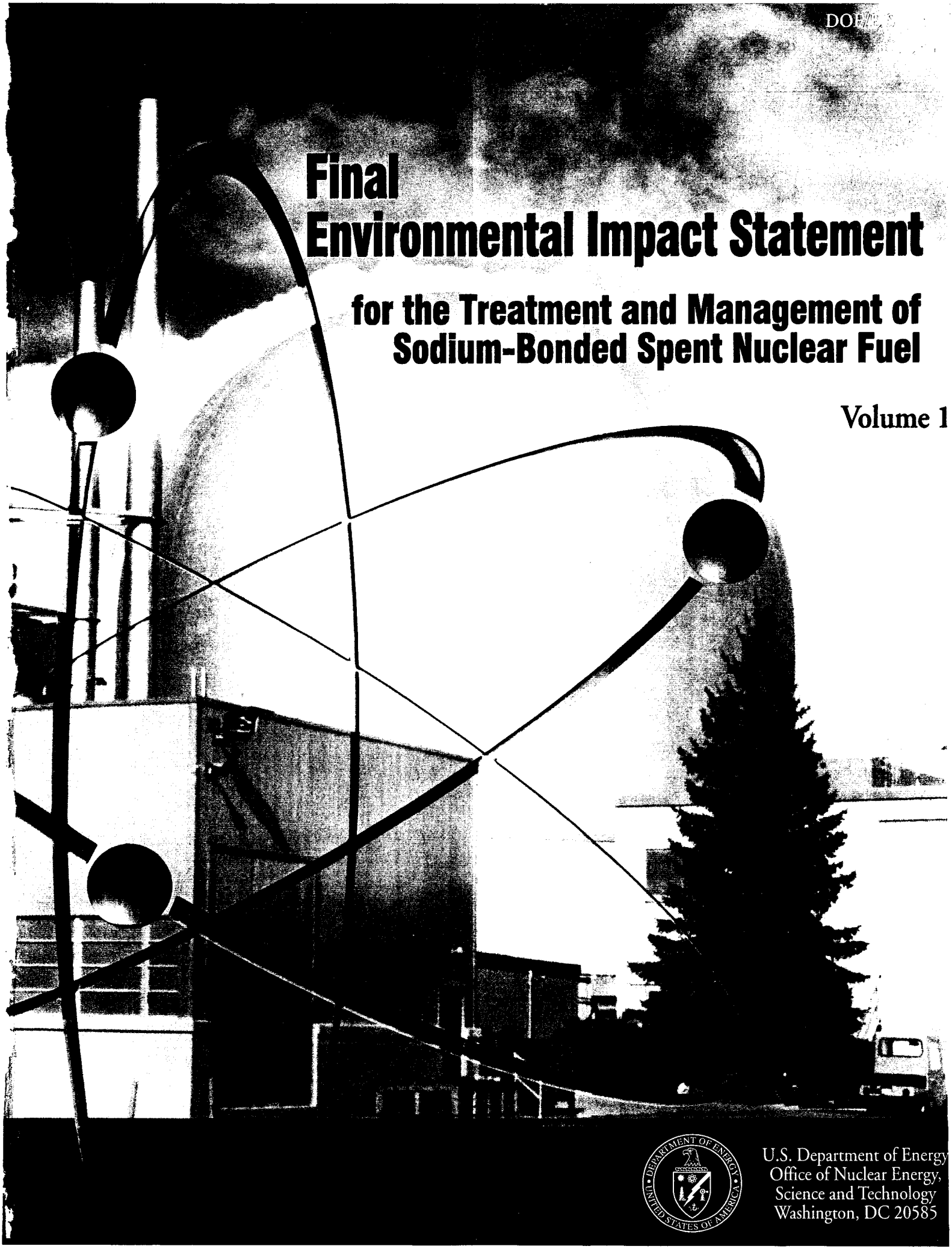
Reprocessing (of spent nuclear fuel) — Processing of reactor-irradiated nuclear material (primarily spent nuclear fuel) to recover fissile and fertile material, to recycle such materials primarily for defense programs. Historically, reprocessing has involved aqueous chemical separation of elements (typically uranium or plutonium) from undesired elements in the fuel.

Roentgen Equivalent Man (rem) — A measure of radiation dose (e.g., the average background radiation dose is 0.3 rem per year). The unit of biological dose equal to the product of the absorbed dose in rads; a quality factor, which accounts for the variation in biological effectiveness of different types of radiation, and other modifying factors.

Spent Nuclear Fuel — Fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated for reprocessing.

Transuranic Waste — Waste contaminated with alpha-emitting radionuclides with half-lives greater than 20 years and concentrations greater than 100 nanocuries/gram at time of assay. It is not a mixed waste.

Treatment — In this EIS, a process to remove and/or stabilize metallic sodium.



Final Environmental Impact Statement

for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel

Volume 1

U.S. Department of Energy
Office of Nuclear Energy, Science and Technology
Washington, DC 20585

In response to comments on the SBSNF Draft EIS and as a result of information that was unavailable at the time of the issuance of the draft EIS, the final EIS contains revisions and new information. These revisions and new information are indicated by a double underline for minor word changes or by a sidebar in the margin for sentence or larger additions. Appendix A contains the comments received during the public review period of the SBSNF Draft EIS and DOE's responses to these comments.

AVAILABILITY OF THE FINAL SBSNF EIS

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Abstract: DOE is responsible for the safe and efficient management of its sodium-bonded spent nuclear fuel. This fuel contains metallic sodium, a highly reactive material; metallic uranium, which is also reactive; and in some cases, highly enriched uranium. The presence of reactive materials could complicate the process of qualifying and licensing DOE's sodium-bonded spent nuclear fuel inventory for disposal in a geologic repository. Currently, more than 98 percent of this inventory is located at the Idaho National Engineering and Environmental Laboratory (INEEL), near Idaho Falls, Idaho. In addition, in a 1995 agreement with the State of Idaho, DOE committed to remove all spent nuclear fuel from Idaho by 2035. This EIS evaluates the potential environmental impacts associated with the treatment and management of sodium-bonded spent nuclear fuel in one or more facilities located at Argonne National Laboratory-West (ANL-W) at INEEL and either the F-Canyon or Building 105-L at the Savannah River Site (SRS) near Aiken, South Carolina. DOE has identified and assessed six proposed action alternatives in this EIS. These are: (1) electrometallurgical treatment of all fuel at ANL-W, (2) direct disposal of blanket fuel in high-integrity cans with the sodium removed at ANL-W, (3) plutonium-uranium extraction (PUREX) processing of blanket fuel at SRS, (4) melt and dilute processing of blanket fuel at ANL-W, (5) melt and dilute processing of blanket fuel at SRS, and (6) melt and dilute processing of all fuel at ANL-W. In addition, Alternatives 2 through 5 include the electrometallurgical treatment of driver fuel at ANL-W. Under the No Action Alternative, the EIS evaluates both the continued storage of sodium-bonded spent nuclear fuel until the development of a new treatment technology or direct disposal without treatment. Under all of the alternatives, the affected environment is primarily within 80 kilometers (50 miles) of spent nuclear fuel treatment facilities. Analyses indicate little difference in the environmental impacts among alternatives. DOE has identified electrometallurgical treatment as its Preferred Alternative for the treatment and management of all sodium-bonded spent nuclear fuel, except for the Fermi-1 blanket fuel. The No Action Alternative is preferred for the Fermi-1 blanket spent nuclear fuel.

Public Comments: The draft EIS was issued for public review and comment on July 31, 1999. The comment period ended on September 28, 1999, although late comments were accepted. Public hearings to solicit

comments on the draft EIS were held in North Augusta, South Carolina; Boise and Idaho Falls, Idaho; and Arlington, Virginia. All comments were considered during the preparation of the final EIS, which also incorporates additional and new information received since the issuance of the draft EIS. In response to comments on the SBSNF Draft EIS and as a result of information that was unavailable at the time of the issuance of the draft EIS, the final EIS contains revisions and new information. These revisions and new information are indicated by a double underline for minor word changes or by a sidebar in the margin for sentence or larger additions. Appendix A contains the comments received during the public review period of the SBSNF Draft EIS and DOE's responses to these comments. DOE will use the analyses presented in this final EIS as well as other information in preparing the Record of Decision for the treatment and management of its sodium-bonded spent nuclear fuel. DOE will issue this Record of Decision no sooner than 30 days after the U.S. Environmental Protection Agency publishes a notice of availability of this final EIS in the *Federal Register*.

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Acronyms, Abbreviations, and Conversion Charts

ACRONYMS, ABBREVIATIONS, AND CONVERSION CHARTS

ANL-W	Argonne National Laboratory-West
BEIR	Biological Effects of Ionizing Radiation
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CFR	Code of Federal Regulations
CPP	Chemical Processing Plant
DOE	U.S. Department of Energy
EBR-II	Experimental Breeder Reactor-II
EIS	Environmental Impact Statement
EPA	U.S. Environmental Protection Agency
ERPG	Emergency Response Planning Guideline
FR	*Federal Register*
GMODS	Glass Material Oxidation and Dissolution System
HFEF	Hot Fuel Examination Facility
INEEL	Idaho National Engineering and Environmental Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
MEDEC	melt, drain, evaporate, and calcine (ANL-W process)
NAAQS	National Ambient Air Quality Standards
NEPA	National Environmental Policy Act
NPDES	National Pollutant Discharge Elimination System
NRC	U.S. Nuclear Regulatory Commission
OSHA	Occupational Safety and Health Administration
PM_n	Particulate matter less than or equal to *n* microns in diameter
PUREX	Plutonium-Uranium Extraction
RCRA	Resource Conservation and Recovery Act
SBSNF	Sodium-Bonded Spent Nuclear Fuel
SRS	Savannah River Site
U.S.C.	United States Code
YAG	yttrium-aluminum-garnet

Metric Conversion Chart

To Convert Into Metric			*To Convert From Metric*		
If You Know	**Multiply By**	**To Get**	**If You Know**	**Multiply By**	**To Get**
Length					
inches	2.54	centimeters	centimeters	0.3937	inches
feet	30.48	centimeters	centimeters	0.0328	feet
feet	0.3048	meters	meters	3.281	feet
yards	0.9144	meters	meters	1.0936	yards
miles	1.60934	kilometers	kilometers	0.6214	miles
Area					
square inches	6.4516	square centimeters	square centimeters	0.155	square inches
square feet	0.092903	square meters	square meters	10.7639	square feet
square yards	0.8361	square meters	square meters	1.196	square yards
acres	0.40469	hectares	hectares	2.471	acres
square miles	2.58999	square kilometers	square kilometers	0.3861	square miles
Volume					
fluid ounces	29.574	milliliters	milliliters	0.0338	fluid ounces
gallons	3.7854	liters	liters	0.26417	gallons
cubic feet	0.028317	cubic meters	cubic meters	35.315	cubic feet
cubic yards	0.76455	cubic meters	cubic meters	1.308	cubic yards
Weight					
ounces	28.3495	grams	grams	0.03527	ounces
pounds	0.4536	kilograms	kilograms	2.2046	pounds
short tons	0.90718	metric tons	metric tons	1.1023	short tons
Temperature					
Fahrenheit	Subtract 32, then multiply by 0.55556	Celsius	Celsius	Multiply by 1.8, then add 32	Fahrenheit

Metric Prefixes

Prefix	*Symbol*	*Multiplication Factor*
exa-	E	1 000 000 000 000 000 000 = 10^{18}
peta-	P	1 000 000 000 000 000 = 10^{15}
tera-	T	1 000 000 000 000 = 10^{12}
giga-	G	1 000 000 000 = 10^{9}
mega-	M	1 000 000 = 10^{6}
kilo-	k	1 000 = 10^{3}
hecto-	h	100 = 10^{2}
deka-	da	10 = 10^{1}
deci-	d	0.1 = 10^{-1}
centi-	c	0.01 = 10^{-2}
milli-	m	0.001 = 10^{-3}
micro-	μ	0.000 001 = 10^{-6}
nano-	n	0.000 000 001 = 10^{-9}
pico-	p	0.000 000 000 001 = 10^{-12}
femto-	f	0.000 000 000 000 001 = 10^{-15}
atto-	a	0.000 000 000 000 000 001 = 10^{-18}

Chapter 1

Introduction

1. INTRODUCTION

Chapter 1 provides an overview of the U.S. Department of Energy's proposal for treatment and management of sodium-bonded spent nuclear fuel. This chapter discusses the background, purpose and need for agency action, and scope of the *Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel*. Included are discussions on the decisions to be made and issues identified by the public during the scoping and public comment periods. The chapter concludes with sections on the relationship of this proposal to other actions and programs under the National Environmental Policy Act and the organization of the document.

1.1 BACKGROUND

For nearly four decades, research, development, and demonstration activities associated with liquid metal fast breeder reactors were conducted at the Experimental Breeder Reactor-II (EBR-II), about 40 miles west of Idaho Falls, Idaho; the Enrico Fermi Atomic Power Plant[1] in Monroe, Michigan; and the Fast Flux Test Facility at the Hanford site in Richland, Washington. These activities generated approximately 60 metric tons of heavy metal of sodium-bonded spent nuclear fuel for which the U.S. Department of Energy (DOE) is now responsible. Sodium-bonded spent nuclear fuel is distinguished from commercial nuclear reactor spent nuclear fuel by the presence of metallic sodium, a highly reactive material; frequently by metallic uranium and plutonium, which are also potentially reactive; and in some cases, highly enriched uranium. Metallic sodium in particular presents challenges for management and ultimate disposal of this spent nuclear fuel. For example, metallic sodium reacts with water to produce explosive hydrogen gas and corrosive sodium hydroxide; both could affect operation of a geologic repository.

DOE proposes to resolve this problem by treating and managing the sodium-bonded spent nuclear fuel and facilitating its ultimate disposal in a geologic repository. The reasonable alternatives for this proposed action are determined by the technology options available to DOE. Several technologies that might be used to treat and manage DOE's sodium-bonded spent nuclear fuel are at various stages of development. Among these are: an electrometallurgical treatment process; the plutonium-uranium extraction (PUREX) process; placement of the spent nuclear fuel in high-integrity cans; a melt and dilute process; a glass material oxidation and dissolution system (GMODS) process; a direct plasma arc-vitreous ceramic process; and a chloride volatility process.

The programmatic risk in implementing any of these potential alternatives for treatment and management of sodium-bonded spent nuclear fuel, or of not treating this fuel, is the uncertainty surrounding the acceptability of DOE spent nuclear fuel for placement in a potential geologic repository. While DOE has drafted preliminary waste acceptance criteria for a geologic repository (DOE 1999c), the final acceptance criteria will be more refined. If the proposed repository at Yucca Mountain in Nevada is developed, final acceptance criteria would not be available until about 2005, when the U.S. Nuclear Regulatory Commission (NRC) would issue a construction authorization. Until such time, the preliminary acceptance criteria are intended to be conservative to allow for uncertainties in the performance of engineered and natural barriers and how such performance might impact public and worker health and safety, as well as material isolation.

This environmental impact statement (EIS) follows the June 1, 1995, Record of Decision (60 FR 28680) for the *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact*

[1] *The original name of the plant was the Enrico Fermi Atomic Power Plant. The numeral "1" was added to the name in 1969 after Detroit Edison Company began construction of Fermi-2. The plant also is known as Fermi, Fermi-1, or Enrico Fermi-1.*

Statement (Programmatic Spent Nuclear Fuel EIS) (DOE 1995a), in which DOE decided to regionalize spent nuclear fuel management by fuel type for DOE-owned spent nuclear fuel. DOE also decided to: (1) continue environmental restoration activities at the Idaho National Engineering and Environmental Laboratory (INEEL)[2]; (2) develop cost-effective treatment technologies for spent nuclear fuel and waste management; and (3) implement projects and facilities to prepare waste and treat spent nuclear fuel for interim storage and final disposition. The 1995 Record of Decision was based partially on the analyses in the Programmatic Spent Nuclear Fuel EIS, which analyzed the potential environmental consequences of alternatives for transporting, receiving, processing, and storing spent nuclear fuel under DOE's responsibility for the next 40 years. The Programmatic Spent Nuclear Fuel EIS also analyzed the consequences of 10 years of waste and spent nuclear fuel management and environmental restoration actions at INEEL.

In addition, DOE committed to remove all spent nuclear fuel from Idaho by 2035 in a 1995 agreement with the State of Idaho (Settlement Agreement and Consent Order [Idaho 1995] issued on October 17, 1995, in the actions of *Public Service Co. of Colorado v. Batt*, No. CV 91-0035-S-EJL [D. Id.], and *United States v. Batt*, No. CV 91-0054-EJL [D. Id.]). Currently, more than 98 percent of DOE's sodium-bonded spent nuclear fuel is located at INEEL near Idaho Falls, Idaho, and is subject to the requirements of this Settlement Agreement and Consent Order. Before sodium-bonded spent nuclear fuel can be removed from the State of Idaho for ultimate disposal, some or all of the fuel may require treatment.

One of the technologies considered for the treatment of sodium-bonded spent nuclear fuel is the electrometallurgical technology. In a 1995 report (NAS 1995), the National Academy of Sciences' National Research Council Committee on Electrometallurgical Techniques for DOE Spent Nuclear Fuel Treatment recommended that DOE confirm the technical feasibility and cost-effectiveness of electrometallurgical treatment of its sodium-bonded spent nuclear fuel. The Council recommended this be done through a technology demonstration using sodium-bonded spent nuclear fuel that had been removed from EBR-II at Argonne National Laboratory-West (ANL-W). Prior to acting on the recommendation, DOE prepared the *Environmental Assessment for the Electrometallurgical Treatment Research and Demonstration Project in the Fuel Conditioning Facility at Argonne National Laboratory-West* (DOE 1996a) and issued a Finding of No Significant Impact on May 22, 1996 (61 FR 25647). The Electrometallurgical Treatment Research and Demonstration Project, which began in June 1996, involved the treatment of up to 100 EBR-II driver spent nuclear fuel assemblies and up to 25 EBR-II blanket spent nuclear fuel assemblies (approximately 1.6 metric tons of heavy metal). The driver spent nuclear fuel contained highly enriched uranium and was used in the active region of the nuclear reactor core. The blanket spent nuclear fuel contained depleted uranium and was used in areas around and near the driver spent nuclear fuel in the reactor core. The Electrometallurgical Treatment Research and Demonstration Project was successfully completed in August 1999. The key analytical and experimental results of the demonstration project are provided in the *Spent Fuel Treatment Demonstration Final Report* issued by ANL-W (Benedict et al. 1999) in August 1999. The salient features of the demonstration project and results are discussed in Section 1.6.3.

Parallel to the assessment provided in this *Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel* (SBSNF EIS), the National Research Council is continuing to evaluate the Electrometallurgical Treatment Research and Demonstration Project. In its most recent report, *Electrometallurgical Techniques for U.S. Department of Energy Spent Fuel Treatment–Spring 1998 Status Report on Argonne National Laboratory's R&D Activity* (NAS 1998), the Council acknowledged progress in the demonstration and recommended that it be carried to completion. Data from the demonstration project were used in preparing this EIS. The National Research Council issued a final report on the technology demonstration in April 2000. DOE will consider the Council's final report in reaching a decision regarding the disposition of sodium-bonded spent nuclear fuel.

[2] *The laboratory's name was changed from Idaho National Engineering Laboratory to Idaho National Engineering and Environmental Laboratory in January 1997.*

1.2 PURPOSE AND NEED FOR ACTION

Sodium-bonded spent nuclear fuel contains metallic sodium. The presence of metallic sodium in the sodium-bonded spent nuclear fuel could potentially complicate disposal certification and licensing for the ultimate disposal of this spent nuclear fuel in a geologic repository. Metallic sodium reacts vigorously with water, producing heat, potentially explosive hydrogen gas, and sodium hydroxide, a corrosive substance. Sodium also is pyrophoric (i.e., a material that is susceptible to spontaneous ignition and continuous combustion). Sodium metal was used as a heat-transfer medium within the stainless steel cladding (outer layer) of the nuclear fuel and as a coolant in the nuclear reactors that used this fuel. To the extent possible, sodium was removed from the external surface of this fuel after its use, but a portion remains bonded to the uranium metal alloy fuel within the cladding and cannot be removed without further treatment. Most (i.e., 99 percent by weight) of the sodium-bonded spent nuclear fuel contains metallic uranium and plutonium. Some metals, such as pure uranium and pure plutonium, are reactive in the presence of air and moisture. The current preliminary repository waste acceptance criteria (DOE 1999c) exclude reactive and potentially explosive materials from being accepted into a geologic repository unless they exist only in trace quantities. Additionally, some of the sodium-bonded spent nuclear fuel contains highly enriched uranium that could create criticality concerns requiring control methods.

To ensure that the terms of the State of Idaho Settlement Agreement and Consent Order are met, and to facilitate disposal, DOE needs to reduce the uncertainties associated with qualifying sodium-bonded spent nuclear fuel for disposal. Appropriate treatment and management of the sodium-bonded spent nuclear fuel would significantly reduce complications related to disposal qualification. Technologies for spent nuclear fuel treatment that could facilitate such qualification therefore should be considered in reaching a decision for treatment of DOE-owned sodium-bonded spent nuclear fuel. Several treatment technologies are at various stages of development and could be used to remove and stabilize the metallic sodium and immobilize or isolate the transuranic and fission products that are in the sodium-bonded spent nuclear fuel. Such technologies include the electrometallurgical treatment process; the PUREX process; placement of the spent nuclear fuel in high-integrity cans; a melt and dilute process; the GMODS process; a direct plasma arc-vitreous ceramic process; and a chloride volatility process.

It is prudent to evaluate these alternative treatment technologies now, while DOE is performing site characterization activities for a potential geologic repository at Yucca Mountain, Nye County, Nevada. Potential waste forms resulting from treatment or packaging of sodium-bonded spent nuclear fuel should be developed as much as possible in parallel with any repository development. The process of establishing a repository depends on not only the site but also the materials for disposal. As part of this process, a total system performance assessment that describes the probable behavior of the repository is performed. This total system assessment includes the performance of the specific waste forms and inventories proposed for disposal. As part of the process of establishing a repository, data for the waste forms are needed prior to making a final repository selection.

Having completed the Electrometallurgical Treatment Research and Demonstration Project (see Section 1.6.3) and in considering the future of PUREX processing capabilities, DOE now needs to decide whether these technologies are suitable for treating the remaining sodium-bonded spent nuclear fuel, or whether there is sufficient reason to delay a decision and wait for the development of other treatment technologies. Delaying this NEPA process could result in a loss of capability and of technical staff knowledgeable about and experienced with the demonstration project. This was an important consideration in the decision to proceed with this EIS.

1.3 PUBLIC PARTICIPATION

1.3.1 Issues Identified During the Scoping Period

On February 22, 1999, DOE published in the *Federal Register* a Notice of Intent to prepare an *Environmental Impact Statement for Electrometallurgical Treatment of Sodium-Bonded Spent Nuclear Fuel in the Fuel Conditioning Facility at Argonne National Laboratory-West* (64 FR 8553). In this Notice of Intent, DOE invited the public to participate and comment on the proposed scope of the EIS. Subsequent to this notice, DOE held four public scoping meetings. The first meeting was attended by about 60 persons and was held in Idaho Falls, Idaho, on March 9, 1999. The second meeting was held in Boise, Idaho, on March 11, 1999, and was attended by 7 persons. Ten persons attended the third meeting, which was held in North Augusta, South Carolina, on March 15, 1999. The fourth meeting was held in Arlington, Virginia, on March 18, 1999, and was attended by 8 persons. A court reporter recorded oral comments at each of these meetings. Written statements or comments from the public also were collected at the meetings. In addition, the public was invited to send comments to DOE by letter, e-mail via the Internet, a toll-free telephone number, and facsimile. The public scoping comment period began with the publication of the Notice of Intent in the *Federal Register* on February 22, 1999 (64 FR 8553), and ended 45 days later on April 8, 1999.

A total of 228 comments were received during the public scoping comment period. All comments were reviewed and considered by DOE in developing the scope of this EIS. A summary of scoping comments and their disposition is provided in Appendix A of this EIS. The significant issues identified during the public scoping period are addressed below.

Many commentors at the public scoping meetings asked specific, technical questions about the proposed action. Areas of interest included:

Waste volume reduction
Nature of the spent nuclear fuel at ANL-W
Waste forms characterization
Waste disposition and qualification (repository acceptance criteria)
PUREX process
Use of facilities
Nonproliferation impacts
Transportation
Demonstration project

A number of persons commented on the schedule for this EIS. Many stated that the draft EIS should not be issued for public comment before publication of other related reports, such as the National Research Council's Waste Qualification Assessment and the National Academy of Sciences' Independent Assessment Final Report on the Electrometallurgical Treatment Research and Demonstration Project; a Nonproliferation Impacts Assessment; and an independent Cost Study. Several commentors said that this EIS is premature because the demonstration project will not be completed until after the draft EIS is published.

Several commentors asked that the EIS include information about the costs of the proposed action and all of the technology alternatives under consideration. Other commentors stated that the public should have an opportunity to comment on the Nonproliferation Impacts Assessment in the same time frame as the draft EIS, or that this EIS should be delayed until the Nonproliferation Impacts Assessment becomes publicly available. Some suggested that the Nonproliferation Impacts Assessment be included in the EIS. A few commentors expressed the opinion that electrometallurgical treatment of spent nuclear fuel is a proliferation-prone technology.

Many waste-related comments included opinions about whether low-enriched uranium, plutonium, noble metals, and other components of the waste stream should be viewed as waste or potentially valuable resources. Several commentors asked that the EIS clarify which specific waste forms would be generated by the treatment processes. Others said the EIS should clarify whether the waste would remain at the Savannah River Site (SRS) after processing or be returned to Idaho if the PUREX process were used. Some commentors argued that the electrometallurgical treatment alternative would not reduce the volume of waste to be stored in a repository. A few questioned how DOE can ensure the waste will meet the acceptance criteria for a repository when no one knows what those criteria will be—or if there will be any repository at all. A few others recommended that the EIS evaluate the PUREX process before it is shut down to ensure that the waste forms resulting from electrometallurgical treatment are as good as the borosilicate glass that is being prepared for a geologic repository.

The commentors generally agreed that DOE should evaluate in detail all of the alternative treatment technologies that potentially could meet DOE's treatment and management needs, even those that DOE considers less technologically mature. Several commentors expressed the opinion that DOE already has made a technology decision in favor of electrometallurgical treatment, but that other alternative new technologies should not be dismissed because of a lack of knowledge about them. Some asked that the EIS: (1) explain how DOE can consider the PUREX process a reasonable alternative when, historically, it could not handle sodium-bonded spent nuclear fuel, and (2) evaluate whether changes in the PUREX process would be needed to accommodate sodium-bonded spent nuclear fuel. A few commentors suggested the EIS should analyze blanket and driver spent nuclear fuel separately, since they have different chemical and radiological characteristics and different treatments might be warranted.

Comments concerning environment, safety, and health issues were comparatively few, as were comments about transportation safety and security.

Comments received during the scoping period were systematically reviewed and evaluated to determine whether the issues raised fell within the scope of the EIS. The comments are addressed in the EIS as indicated in Appendix A, Table A–1, which includes references to specific EIS sections. As a result of public comment, DOE changed the proposed action of the EIS, as well as the structure of the alternatives. The proposed action was changed from electrometallurgical treatment of sodium-bonded spent nuclear fuel in the Fuel Conditioning Facility at ANL-W to the treatment and management of sodium-bonded spent nuclear fuel. The title also was changed accordingly. This change was made to alleviate concerns about bias for one treatment technology over others. The alternatives were restructured to reflect differences in the characteristics of the different types of sodium-bonded spent nuclear fuel. Thus, several alternatives were added that treat driver and blanket spent nuclear fuel by different technologies.

Issues related to cost and nuclear nonproliferation were not considered to be within the scope of the EIS. However, DOE conducted a Cost Study and a Nonproliferation Impacts Assessment for the reasonable alternatives. These reports were made available to the public during the public review process.

With respect to comments related to the ongoing Electrometallurgical Treatment Research and Demonstration Project, data from the project were used for the preparation of both the draft and the final EIS as indicated in Section 1.6.3.

Comments considered to be not within the scope of the EIS are listed in Appendix A, Table A–3, along with an explanation for their disposition.

1.3.2 Issues Raised During the Public Comment Period on the Draft EIS

In July 1999, DOE published the *Draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel.* The regulations implementing the National Environmental Policy Act (NEPA) mandate a minimum 45-day public comment period after publication of a draft EIS to provide an opportunity for the public and other stakeholders to comment on the EIS analysis and results. The 45-day public comment period on the SBSNF Draft EIS began on July 31, 1999, and was scheduled to end on September 13, 1999. In response to commentor requests, the comment period was extended an additional 15 days through September 28, 1999. During this 60-day comment period, public hearings were held in North Augusta, South Carolina; Boise and Idaho Falls, Idaho; and Arlington, Virginia. In addition, the public was encouraged to submit comments via the U.S. mail service, electronic mail, a toll-free 800-number phone line, and a toll-free fax line.

A total of 494 comments were received during the public comment period. Most of the comments focused on the following issues: (1) the purpose, need for, and timing of the proposed action; (2) the introduction of new waste forms produced by the proposed action, their acceptability in a geologic repository, and the disposition of uranium and plutonium by-products; (3) the public availability of information considered relevant to reviewing the draft EIS, the extension of the comment period, and the relationship of the EIS to other DOE programs; (4) the cost of the various alternatives; (5) the impacts of the proposed action on U.S. nuclear nonproliferation policy; (6) technical or NEPA-related questions regarding technologies and alternatives; and (7) questions related to the affected environment and the environmental consequences. DOE's responses to these issues are summarized below. The comments also dealt with a number of other subjects, including technologies considered and dismissed from further evaluation, long-term (beyond institutional control) performance of the sodium-bonded spent nuclear fuel during storage on site, and questions on the methodology and assumptions of the health and safety analysis. Many commentors expressed their opposition or support for DOE's action in general or for specific alternatives under the proposed action or the No Action Alternative. Section A.2 of Appendix A provides DOE's responses to all comments on a comment-by-comment basis.

Purpose, Need for, and Timing of the Proposed Action

Many comments expressed the opinion that DOE failed to demonstrate the purpose and need for the proposed action or to provide a rationale for its timing. Some of the reasons given included the lack of a compelling argument that there is a safety risk associated with current storage; the lack of a regulatory framework and final waste acceptance criteria; the lack of an approved site for a geologic repository; insufficient information on the results of the Electrometallurgical Treatment Research and Demonstration Project; and the lack of analysis showing that direct disposal of the sodium-bonded spent nuclear fuel without sodium removal would be detrimental to the performance of the geologic repository.

DOE's position, presented in the EIS, is that the need to examine options for the treatment and management of sodium-bonded spent nuclear fuel is based on the existing regulatory environment concerning long-term disposal of spent nuclear fuel and high-level radioactive waste. DOE assumes that its sodium-bonded spent nuclear fuel, as well as other DOE-owned spent nuclear fuel, eventually will be disposed of in a geologic repository. However, one of the key requirements, as specified in the current April 1999 version of DOE's Waste Acceptance Systems Requirements Document (DOE 1999c) and in NRC requirements for acceptance of spent nuclear fuel or high-level radioactive waste in a geologic repository, is that it cannot contain or generate materials that are explosive, pyrophoric, or chemically reactive in a form or amount that could compromise the repository's ability to perform its waste isolation function or to satisfy its performance objective [10 CFR 60.135(b)(1)]. The sodium-bonded spent nuclear fuel, if left in its existing state, would contain pyrophoric and chemically reactive metallic sodium and therefore may not meet DOE or NRC repository acceptance criteria, or would complicate the qualification process.

The timing for the proposed action is a programmatic issue rather than a safety issue. The EIS does not conclude that current storage of sodium-bonded spent nuclear fuel presents a threat to the health and safety of workers or the public. The programmatic risk associated with implementing the proposed action or not treating the sodium-bonded spent nuclear fuel is the uncertainty surrounding the acceptability of this fuel for placement in a geologic repository. The process of establishing a repository is dependent on not only the site but also the materials to be disposed. As part of this process, a total system performance assessment that describes the probable behavior of a repository is performed. This total system assessment includes the performance of the specific waste forms and inventories proposed for disposal. As part of the process of establishing a repository, data for the waste forms are needed prior to making a final repository selection, not after. In fact, if specific waste forms are not represented in crucial documents like this EIS, new documentation will be needed to allow the possibility of disposing of those materials in the repository. The performance of sodium-bonded spent nuclear fuel in a geologic repository depends on many factors (e.g., long-term fuel integrity and fuel/waste package survivability in a repository environment), and the presence of metallic sodium would complicate the modeling even further. Stabilization of the sodium-bonded spent nuclear fuel and/or removal of the metallic sodium would provide greater protection for human health and the environment.

The Electrometallurgical Treatment Research and Demonstration Project began in June 1996 and, although the review of the test results has not been finalized in a single report, a number of status reports were issued by DOE and reviewed by the National Academy of Sciences' National Research Council Committee. They are referenced in the EIS. The success criteria established at the outset of the project have been fulfilled. The environmental impact analysis associated with the electrometallurgical treatment process alternatives was based on actual data from the demonstration project. This final EIS includes a new section on the status and results of the project. Having completed the demonstration project and in considering the future of its PUREX processing capabilities, DOE now needs to decide whether these technologies are suitable for treating the remaining sodium-bonded spent nuclear fuel, or whether there is sufficient reason to delay a decision and wait for the development of other treatment technologies. Delaying the NEPA process could result in a loss of capability and of technical staff knowledgeable about and experienced with the demonstration project. This was an important consideration in the decision to proceed with this EIS.

New Waste Forms and Disposition of Uranium and Plutonium By-Products

Some of the comments questioned the generation of new waste forms from treating the sodium-bonded spent nuclear fuel and the possible acceptance of these forms in a geologic repository. Also, a number of commentors remarked on the generation of uranium and plutonium as by-products of the treatment process. Related issues were the disposition of uranium metal, a by-product of the electrometallurgical process, and the compliance of both the PUREX and the electrometallurgical processes with U.S. nuclear nonproliferation policy in terms of the separation of these elements.

All of the alternatives evaluated in this EIS would produce some form of high-level radioactive waste. Electrometallurgical treatment would produce two new waste forms (i.e., metallic and ceramic) and the melt and dilute process would produce a new metallic form (i.e., a melt and dilute product, or conditioned spent nuclear fuel). These forms would be more stable than the untreated sodium-bonded spent nuclear fuel. The production of a chemically stable waste form to replace a chemically reactive waste form (i.e., sodium-bonded spent nuclear fuel) represents an improvement in the safe, long-term storage of this spent nuclear fuel. DOE expects the new waste forms to be suitable for disposal in a repository and to meet the requirements of the final waste acceptance criteria. The high-level radioactive waste form resulting from the PUREX process is borosilicate glass, which has been tested and analyzed extensively under conditions relevant to a geologic repository.

With respect to uranium and plutonium disposition, the EIS states that any uranium that would be separated under the electrometallurgical process would be blended down and stored on site if it originates from driver spent nuclear fuel, or would be stored on site as depleted uranium if it originates from blanket spent nuclear fuel. The final disposition of the stored uranium has not been decided and is not discussed in the EIS. The disposition of the uranium will be subject to a separate NEPA review. The nuclear nonproliferation policy aspects of this separation are subject to the nuclear nonproliferation policy assessment of the alternatives. The approximately 260 kilograms (572 pounds) of plutonium that would be separated under the PUREX process would be disposed of in accordance with the Record of Decision (65 FR 1608) for the *Surplus Plutonium Disposition Final Environmental Impact Statement* (DOE 1999e) issued in November 1999. This separation is the subject of the Nonproliferation Impacts Assessment, which is independent of this EIS.

Public Availability of Information and Related Documentation

Many commentors asked for a 60-day extension of the 45-day public comment period on the draft EIS. Commentors said they wanted additional time to obtain and review relevant documents such as the Yucca Mountain Draft EIS and the National Academy of Sciences' National Research Council's final report on the Electrometallurgical Treatment Research and Demonstration Project, as well as the Cost Study and the Nonproliferation Impacts Assessment. The comments frequently stated that DOE needs to make all of this information publicly available before the end of the EIS comment period and the issuance of the final EIS and the Record of Decision.

In an effort to ensure that all interested parties had time to comment on the draft EIS, the due date for transmittal of comments was extended from September 13 to September 28, 1999 (64 FR 49169). With respect to the need for more information, DOE made that information available to the public. Background materials were placed in public reading rooms and were made available to the public through a series of hearings held August 17, 1999, in North Augusta, South Carolina; August 24, 1999, in Boise, Idaho; August 26, 1999, in Idaho Falls, Idaho; and August 31, 1999, in Arlington, Virginia. Materials placed in the reading rooms included the electrometallurgical demonstration environmental assessment, the Finding of No Significant Impact for the environmental assessment, National Research Council reports, the 1995 Settlement Agreement and Consent Order with the State of Idaho, the scoping meeting transcripts and comments, and the draft EIS hearing presentations and fact sheets. In addition, completion of the Cost Study and Nonproliferation Impacts Assessment was expedited so that they would be available to the public during the comment period. These reports were mailed to interested parties on August 12, 1999, and were made available to attendees at all of the public hearings on the draft EIS. Although these reports are not critical to the evaluation of the analysis presented in the draft EIS, they will provide input to the Record of Decision. While the final National Research Council report on the demonstration project was published in April 2000, interim status reports were produced throughout the project. Data generated during the demonstration project were used in preparing the EIS.

Cost Issues

A number of commentors raised cost issues and provided comments directly related to the Cost Study, which is not part of the EIS.

Comments concerning the costs of the proposed action were considered beyond the scope of the EIS. The EIS was prepared in accordance with NEPA, as well as the Council on Environmental Quality's regulations on implementing NEPA (40 CFR 1500 through 1508) and DOE's NEPA regulations (10 CFR 1021). None of these regulations require the inclusion of a cost analysis in an EIS. The basic objective of the SBSNF EIS is to provide the public and DOE decision-makers with a description of the reasonable alternatives for treating and managing sodium-bonded spent nuclear fuel and information about their potential impacts on public health and safety and the environment. While cost could be an important factor in the ultimate Record of Decision,

the purpose of this EIS is to address the environmental consequences of all alternatives under the proposed action and the No Action Alternative. DOE distributed cost information through the independent Cost Study released in August 1999, and this information is available to the public on request and in the DOE public reading rooms. Responses to specific comments related to cost issues are included in Section A.2 of Appendix A.

Nuclear Nonproliferation Policy Issues

The nuclear nonproliferation implications of the proposed action were the subject of a number of comments. Some commentors expressed strong opinions about how the use of specific technologies such as electrometallurgical treatment might impact U.S. nonproliferation policy.

Nonproliferation is another issue that was considered beyond the scope of the EIS. A separate Nonproliferation Impacts Assessment was prepared by DOE's Office of Arms Control and Nonproliferation. After assessing the potential nonproliferation impacts that could result from each of the alternatives and technologies analyzed in the SBSNF Draft EIS, the Office of Arms Control and Nonproliferation found that all the alternatives, except that involving PUREX processing at SRS, are fully consistent with U.S. policy concerning reprocessing and nuclear nonproliferation. Electrometallurgical treatment, for example, would not increase national inventories of weapons-usable fissile material because, although highly enriched uranium is an interim product of the process, it would be blended down to low-enriched uranium during treatment. Within the current equipment configuration and design, it is not possible to produce weapons-usable plutonium merely by adjusting the operating parameters. To do this, traditional aqueous processing would be required after electrometallurgical treatment. However, traditional aqueous processing could be used to produce weapons-usable plutonium directly from the spent nuclear fuel, without electrometallurgical treatment, so electrometallurgical treatment itself does not present a special proliferation concern. Responses to specific comments related to nonproliferation are included in Section A.2 of Appendix A.

Technologies, Alternatives

Various comments dealt with technical questions and issues regarding the treatment technologies addressed in the EIS or NEPA-related issues regarding the selected alternatives.

The variety of the issues precludes a summary response. Responses to these questions on a comment-by-comment basis are included in Section A.2 of Appendix A. A number of revisions to the EIS were made as a result of these comments.

Affected Environment and Consequences

A number of comments included questions concerning the description of the affected environment in the SBSNF Draft EIS, and the results of the environmental impact analysis.

As in the case above, responses to these questions on a comment-by-comment basis are included in Section A.2 of Appendix A.

1.4 Scope of this EIS

The EIS evaluates the potential direct, indirect, and cumulative environmental impacts associated with the treatment of sodium-bonded spent nuclear fuel in one or more spent nuclear fuel management facilities. In addition, this EIS evaluates the environmental impacts of the No Action Alternative.

DOE proposes to treat and manage sodium-bonded spent nuclear fuel at one or more of the following spent nuclear fuel management facilities: ANL-W at INEEL and the F-Canyon or Building 105-L at SRS. The impacts from the treatment and management of sodium-bonded spent nuclear fuel at INEEL and SRS and their spent nuclear fuel management facilities are described in this EIS. In addition to the No Action Alternative, the EIS analyzes six reasonable alternatives under the proposed action that employ one or more of the following technology options: electrometallurgical treatment, the PUREX process, packaging in high-integrity cans, and the melt and dilute treatment process. Electrometallurgical treatment at a site other than ANL-W, the GMODS process, the direct plasma arc-vitreous ceramic treatment, and the chloride volatility process were considered and deemed not to be reasonable alternatives under the proposed action.

This EIS analyzes the potential environmental impacts associated with the proposed action, which includes: (1) preparation prior to treatment; (2) treatment and management; (3) transportation; and (4) decontamination and deactivation of equipment that would be installed for the purpose of implementing a specific treatment method. Impacts from the transport to INEEL of sodium-bonded spent nuclear fuel from DOE sites such as the Hanford site in Washington, Sandia National Laboratories in New Mexico, and Oak Ridge National Laboratory in Tennessee are addressed in the Programmatic Spent Nuclear Fuel EIS (DOE 1995a).

The United States does not encourage the civilian use of plutonium and, accordingly, does not itself engage in plutonium reprocessing for either nuclear power or nuclear explosive purposes. However, two of the technologies under the proposed action involve the separation of plutonium (PUREX) and highly enriched uranium (electrometallurgical treatment). To address concerns that treatment of this fuel by chemical separation could encourage reprocessing in other countries, DOE's Office of Nonproliferation and National Security independently evaluated the impacts of each treatment technology on U.S. nonproliferation efforts. The Nonproliferation Impacts Assessment was published at about the same time as the draft EIS.

1.5 Decisions to be Made

Based on the analytical results of this EIS as well as cost, schedule, and nonproliferation considerations, DOE intends to make the following decisions:

- Whether to use an existing, mature technology to treat the sodium-bonded spent nuclear fuel, and if so, which technology should be selected and where it should be implemented.

- Whether to take no action now and wait for further information regarding the potential development of a geologic repository, or promote the development of a less mature or new treatment technology.

The information presented in this EIS, combined with public comments on the draft EIS, the Nonproliferation Impacts Assessment, a Cost Study of the reasonable alternatives, and the National Research Council's final evaluation of the demonstration project, will enable DOE to make a decision regarding treatment and management of the sodium-bonded spent nuclear fuel.

1.6 Relationship to Other Actions and Programs

This section explains the relationship between this EIS and other relevant NEPA documents and programs. Completed NEPA actions are described in Section 1.6.1, ongoing actions are described in Section 1.6.2, and the Electrometallurgical Treatment Research and Demonstration Project is described in Section 1.6.3.

1.6.1 Completed NEPA Actions

1.6.1.1 Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement

This Programmatic Spent Nuclear Fuel EIS (DOE 1995a) analyzed at a programmatic level the potential environmental consequences of alternatives used for 40 years to transport, receive, process, and store spent nuclear fuel under DOE's responsibility. It also analyzed the consequences of 10 years of waste and spent nuclear fuel management and environmental restoration actions at Idaho National Engineering Laboratory (now known as INEEL). For programmatic spent nuclear fuel management, this document analyzed alternatives that included no action, decentralization, regionalization, centralization, and the use of plans that existed in 1992 and 1993 for the management of these materials. For the INEEL, this document analyzed alternatives such as no action, a 10-year plan, and minimum and maximum treatment, storage, and disposal of DOE waste.

Issued in April 1995, the Programmatic Spent Nuclear Fuel EIS was followed by a Record of Decision published in the *Federal Register* on June 1, 1995 (60 FR 28680). In the Record of Decision, DOE decided to regionalize spent nuclear fuel management by fuel type for DOE-owned spent nuclear fuel. DOE also decided to: (1) continue environmental restoration activities at the INEEL; (2) develop cost-effective treatment technologies for spent nuclear fuel and waste management; and (3) implement projects and facilities to prepare waste and treat spent nuclear fuel for interim storage and final disposition. The SBSNF EIS was prepared as a follow-on to this programmatic EIS.

The June 1, 1995, Record of Decision was later amended to reflect the October 16, 1995, Settlement Agreement and Consent Order between DOE, the State of Idaho, and the U.S. Department of the Navy pertaining to spent nuclear fuel shipments into and out of the State of Idaho. The amendment to the Record of Decision was published in the *Federal Register* on March 8, 1996 (61 FR 9441). In this amendment, DOE did not modify or rescind any of the provisions presented in the June 1, 1995, Record of Decision (60 FR 28680), but reduced the number of shipments of spent nuclear fuel into the State of Idaho.

1.6.1.2 Savannah River Site Waste Management Final Environmental Impact Statement

DOE issued this EIS (DOE 1995b) to provide a basis for the selection of a site-wide approach to managing present and future (through 2024) waste generated at SRS. This waste would come from ongoing operations and potential actions, new missions, environmental restoration, and decontamination and decommissioning programs.

The SRS Waste Management EIS includes the treatment of wastewater discharges in the Effluent Treatment Facility, F- and H-Area tank operations and waste removal, and construction and operation of a replacement high-level radioactive waste evaporator in the H-Area tank farm. In addition, it evaluates the Consolidated Incineration Facility for the treatment of mixed waste. The Record of Decision, published in the *Federal Register* on October 30, 1995 (60 FR 55249), stated that DOE will configure its waste management system according to the moderate treatment alternative described in the EIS. The SRS Waste Management EIS evaluates management alternatives for various types of waste that actions proposed in this EIS could generate.

In a Supplemental Record of Decision published in the *Federal Register* on May 19, 1997 (62 FR 27241), DOE decided to take additional measures to further implement the Moderate Treatment Configuration Alternative for mixed waste and transuranic waste. This decision was based on the SRS Waste Management EIS and was consistent with completed negotiations between DOE and the State of South Carolina.

1.6.1.3 Final Environmental Impact Statement, Interim Management of Nuclear Materials

In this EIS (DOE 1995c) DOE evaluated actions to stabilize nuclear materials at SRS that present potential environmental, safety, and health risks in their current storage condition or may present a risk within the next 10 years. As a result, DOE published five decisions from this EIS. In the Record of Decision, published in the *Federal Register* on December 19, 1995 (60 FR 65300), DOE decided to process, blend, and/or vitrify specific amounts of plutonium, uranium, americium, and curium solutions, and spent nuclear fuel down to low enrichments and/or some other form of stable material. The Savannah River Site Interim Management of Nuclear Materials EIS evaluates the treatment and management of spent nuclear fuel and other waste at SRS such as those generated by the proposed actions in the SBSNF EIS.

In the first, second, and third supplements to the Record of Decision, published in the *Federal Register* on February 21, 1996; September 13, 1996; and April 11, 1997, respectively (61 FR 6633, 61 FR 48474, and 62 FR 17790), DOE decided to stabilize additional amounts of spent nuclear fuel and other materials by processing them in the F-Canyon, H-Canyon, and the FB-Line and blending the resulting highly enriched uranium down to low-enriched uranium. DOE then would transfer the resulting nuclear material to the SRS high-level radioactive waste tanks for vitrification in the Defense Waste Processing Facility.

In the fourth supplement to the Record of Decision, published in the *Federal Register* on November 14, 1997 (62 FR 61099), DOE decided to process, store, and vitrify specific amounts of nuclear material in the Defense Waste Processing Facility and to amend the September 13, 1996, supplement to the Record of Decision (61 FR 48474) to address additional amounts of plutonium and neptunium solutions stored at SRS.

1.6.1.4 Environmental Assessment for the Electrometallurgical Treatment Research and Demonstration Project in the Fuel Conditioning Facility at Argonne National Laboratory-West

This NEPA analysis (DOE 1996a) addressed the environmental impacts associated with a research and demonstration project involving the electrometallurgical treatment of up to 100 EBR-II driver spent nuclear fuel assemblies and up to 25 EBR-II blanket spent nuclear fuel assemblies in the Fuel Conditioning Facility at ANL-W. As noted in the environmental assessment, DOE had identified electrometallurgical treatment as a promising technology to treat EBR-II spent nuclear fuel, but an appropriate demonstration was needed to provide DOE with sufficient information to evaluate its technical feasibility. A successful demonstration of the electrometallurgical treatment technology on EBR-II spent nuclear fuel, combined with research and testing of the resulting waste forms, would provide DOE with the information needed to determine whether this treatment technology should be used to treat the remainder of EBR-II spent nuclear fuel and/or other types of spent nuclear fuel. Based on the analysis presented in the environmental assessment, and after consideration of all the comments received from the public, DOE decided to proceed with the proposed demonstration and finalized the environmental assessment on May 15, 1995. DOE also determined that the proposed action did not constitute a major Federal action and would not necessitate the preparation of an EIS. DOE issued a Finding of No Significant Impact, which was published in the *Federal Register* on May 22, 1996 (61 FR 25647).

The electrometallurgical treatment process that was addressed in this environmental assessment is basically the same process that is being evaluated in this EIS. The process involves the dissolution of spent nuclear fuel by the use of an electric current in a molten salt mixture. The only difference between the environmental assessment and this SBSNF EIS is the amount of spent nuclear fuel being considered for treatment.

The Electrometallurgical Treatment Research and Demonstration Project was completed in August 1999. Salient features of the project and results are discussed in Section 1.6.3.

1.6.1.5 Disposition of Surplus Highly Enriched Uranium Final Environmental Impact Statement

DOE prepared this EIS (DOE 1996b) because of the need to move rapidly to neutralize the proliferation threat of surplus highly enriched uranium and to demonstrate to other nations the United States' commitment to nonproliferation. The Highly Enriched Uranium EIS evaluates management alternatives for materials that actions proposed in this EIS could generate.

In the Record of Decision, published in the *Federal Register* on August 5, 1996 (61 FR 40619), DOE stated it would implement a program that will gradually blend as much as 85 percent of the surplus highly enriched uranium to a uranium-235 enrichment level of approximately 4 percent, and will blend the remaining surplus highly enriched uranium down to an enrichment level of about 0.9 percent for disposal as low-level radioactive waste. This will occur over 15 to 20 years. DOE could use different technologies at four potential blending facilities, including SRS and the Oak Ridge Reservation. Blending down highly enriched uranium would affect SRS operations and waste generation.

1.6.1.6 Final Waste Management Programmatic Environmental Impact Statement for Managing Treatment, Storage, and Disposal of Radioactive and Hazardous Waste

This Final Waste Management Programmatic EIS (DOE 1997) examined the potential environmental and cost impacts of strategic management alternatives for managing five types of radioactive and hazardous waste that have resulted and will continue to result from nuclear defense and research activities at a variety of sites around the United States. The five waste types are mixed waste, low-level radioactive waste, transuranic waste, high-level radioactive waste, and hazardous waste. This programmatic EIS provided information on the impacts of various siting alternatives which DOE will use to decide at which sites to locate additional treatment, storage, and disposal capacity for each waste type. This information included the cumulative impacts of combining future siting configurations for the five waste types and the collective impacts of other past, present, and reasonably foreseeable future activities. The programmatic EIS evaluates management and treatment alternatives for various types of waste that actions proposed in this EIS could generate.

The waste management facilities considered for the five waste types were treatment and disposal facilities for mixed waste; treatment and disposal facilities for low-level radioactive waste; treatment and storage facilities for transuranic waste in the event that treatment is required before disposal; storage facilities for treated (vitrified) high-level radioactive waste canisters; and treatment of nonwastewater hazardous waste by DOE and commercial vendors. In addition to the No Action Alternative, which included only existing or approved waste management facilities, the alternatives for each of the five waste type configurations included decentralized, regionalized, and centralized alternatives for operating existing and new waste management facilities. However, the siting, construction, and operation of any new facility at a selected site would not be decided until completion of a site-wide or project-specific environmental review.

DOE has published four decisions from this programmatic EIS. In the first Record of Decision, published in the *Federal Register* on January 23, 1998 (63 FR 3629), DOE decided that each DOE site that currently has or will generate transuranic waste will prepare and store its transuranic waste on site, except for Sandia National Laboratories/New Mexico, which will transfer its transuranic waste to the Los Alamos National Laboratory. Los Alamos National Laboratory will have facilities that are not available or anticipated at Sandia National Laboratories to prepare and store transuranic waste prior to disposal.

In the second Record of Decision, published in the *Federal Register* on August 5, 1998 (63 FR 41810), DOE decided to continue using offsite facilities for the treatment of major portions of the nonwastewater hazardous waste generated at DOE sites. This decision did not involve any transfer of nonwastewater hazardous waste among DOE sites.

In the third Record of Decision, published in the *Federal Register* on August 26, 1999 (64 FR 46661), DOE decided to store immobilized high-level radioactive waste in a final form at the site of generation (Hanford, INEEL, SRS, and the West Valley Demonstration Project) until transfer to a geologic repository for ultimate disposal.

DOE addressed the management and disposal of low-level and mixed radioactive waste in a fourth Record of Decision, published in the *Federal Register* on February 25, 2000 (65 FR 10061). In this Record of Decision, DOE decided to perform minimum treatment of low-level radioactive waste at all sites and continue, to the extent practicable, disposal of onsite low-level radioactive waste at INEEL, Los Alamos National Laboratory, the Oak Ridge Reservation, and SRS. DOE decided to treat mixed low-level radioactive waste at the Hanford site, INEEL, the Oak Ridge Reservation, and SRS, with disposal at the Hanford site and the Nevada Test Site.

1.6.1.7 Advanced Mixed Waste Treatment Project Final Environmental Impact Statement

This EIS (DOE 1999a) assessed the potential environmental impacts associated with four alternatives related to the construction and operation of the Advanced Mixed Waste Treatment Facility at INEEL. The alternatives analyzed were: the No Action Alternative; the proposed action; the Nonthermal Treatment Alternative; and the Treatment and Storage Alternative. The Advanced Mixed Waste Treatment Facility would treat transuranic waste, mixed waste, and alpha-contaminated mixed waste in preparation for disposal. After treatment, transuranic waste would be disposed of at the Waste Isolation Pilot Plant in New Mexico. Mixed waste would be disposed of at an approved disposal facility depending on decisions to be based on DOE's Final Waste Management Programmatic EIS (DOE 1997). Evaluations of impacts on land use; socioeconomics; cultural resources; aesthetic and scenic resources; geology; air resources; water resources; ecological resources; noise; traffic and transportation; occupational and public health and safety; INEEL services; and environmental justice were included in the assessment. The *Advanced Mixed Waste Treatment Project Final Environmental Impact Statement* addresses waste types that could be generated by actions proposed in this EIS.

In the Record of Decision, published in the *Federal Register* on April 7, 1999 (66 FR 16948), DOE decided to proceed with the construction and operation of the Advanced Mixed Waste Treatment Facility. DOE then will treat and prepare for shipment and disposal 65,000 cubic meters (2.3 million cubic feet) of DOE transuranic waste, mixed waste, alpha-contaminated mixed waste currently stored at INEEL. As a result of the decision to complete this facility, DOE also could treat up to 120,000 cubic meters (4.24 million cubic feet) of additional waste from INEEL or other DOE sites for a total of 185,000 cubic meters (6.53 million cubic feet). The Advanced Mixed Waste Treatment Facility will treat waste to meet the Waste Isolation Pilot Plant Waste Acceptance Criteria and applicable requirements of the Toxic Substances Control Act and the Resource Conservation and Recovery Act (RCRA) Land Disposal Restrictions.

In makings its decision, DOE considered several factors, including the environmental analyses reported in the Advanced Mixed Waste Treatment Project Final EIS; estimated costs of the alternatives reported in the Advanced Mixed Waste Treatment Project EIS Alternatives Cost Study; regulatory implications of the alternatives; mission; national policy; and public comments on the Advanced Mixed Waste Treatment Project Draft EIS. This Record of Decision (66 FR 16948) documents DOE's decision to implement the Preferred Alternative, which provides the greatest long-term protection of the environment with small short-term environmental impacts and health risks

1.6.1.8 Surplus Plutonium Disposition Final Environmental Impact Statement

The Surplus Plutonium Disposition EIS (DOE 1999e) was tiered from the *Storage and Disposition of Weapons-Usable Fissile Materials Programmatic Environmental Impact Statement* (DOE 1996c). The Record of Decision for the programmatic EIS, published in the *Federal Register* on January 14, 1997

(62 FR 3014), outlined DOE's approach to plutonium disposition and established the groundwork for the Surplus Plutonium Disposition EIS. The fundamental purpose of the program is to ensure that plutonium produced for nuclear weapons and declared excess to national security needs (now and in the future) will never again be used for nuclear weapons.

The Surplus Plutonium Disposition EIS evaluated reasonable alternatives for the siting, construction, and operation of facilities required to implement DOE's disposition strategy for up to 50 metric tons of surplus plutonium, including a No Action Alternative. The disposition facilities analyzed in this EIS include pit disassembly and conversion, plutonium conversion and immobilization, and mixed oxide fuel fabrication. The Surplus Plutonium Disposition EIS also analyzed the potential impacts of fabricating a limited number of mixed oxide fuel assemblies for testing in a reactor. The Surplus Plutonium Disposition EIS is a related NEPA action because it addresses the disposition of material that the SBSNF EIS could generate.

In the Record of Decision, published in the *Federal Register* on January 11, 2000 (65 FR 1608), DOE decided to provide for the safe and secure disposition of up to 50 metric tons of surplus plutonium by constructing all three disposition facilities, pit disassembly and conversion, plutonium conversion and immobilization, and mixed oxide fuel fabrication, at SRS. DOE also decided to implement the mixed oxide fuel alternative analyzed in the EIS.

1.6.2 Ongoing NEPA Actions

1.6.2.1 Savannah River Site Spent Nuclear Fuel Management Final Environmental Impact Statement

The *Savannah River Site Spent Nuclear Fuel Management Final Environmental Impact Statement* (DOE 2000) was issued in March 2000. This SRS Spent Nuclear Fuel Final EIS analyzed the potential impacts from the management of spent nuclear fuel and targets assigned to SRS, including placing these materials in forms suitable for ultimate disposition. Options to treat, package, and store spent nuclear fuel are discussed in this document. The material addressed by this EIS consists of approximately 68 metric tons of heavy metal of spent nuclear fuel (including 20 metric tons of heavy metal of uranium-thorium spent nuclear fuel at SRS; approximately 28 metric tons of heavy metal of aluminum-clad spent nuclear fuel from foreign and domestic research reactors to be shipped to SRS through 2035; and 20 metric tons of heavy metal of stainless steel or zirconium-clad spent nuclear fuel, as well as some other programmatic material stored at SRS for repackaging and dry storage pending shipment off site).

The alternatives considered in the SRS Spent Nuclear Fuel Final EIS encompass a range of new packaging, new processing, and conventional reprocessing technologies for the treatment of spent nuclear fuel. Many of these technologies also are analyzed in this SBSNF EIS. However, in the SRS Spent Nuclear Fuel Final EIS, DOE chose melt and dilute and conventional processing (PUREX) as preferred treatment alternatives for the spent nuclear fuel assigned to SRS.

1.6.2.2 Draft Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada

This draft EIS (DOE 1999b) assesses the potential environmental impacts from the proposed construction, operation, monitoring, and closure of an NRC-licensed geologic repository for the disposal of spent nuclear fuel and high-level radioactive waste, as mandated by the Nuclear Waste Policy Act, as amended. The Yucca Mountain EIS is required to accompany any DOE site recommendation to the President, as appropriate, under Section 114 of the Nuclear Waste Policy Act.

The proposed action addressed in this EIS is to construct, operate and monitor, and eventually close a geologic repository at Yucca Mountain in southern Nevada for the disposal of spent nuclear fuel and high-level

radioactive waste currently in storage at 72 commercial and 5 DOE sites across the United States. The EIS evaluates (1) projected impacts on the Yucca Mountain environment from the construction, operation and monitoring, and eventual closure of the geologic repository; (2) the potential long-term impacts of repository disposal of spent nuclear fuel and high-level radioactive waste; (3) the potential impacts of transporting these materials nationally and in the state of Nevada; and (4) the potential impacts of not proceeding with the proposed action. Included in the high-level radioactive waste that is assumed to be disposed of at the repository are the metallic and ceramic waste forms that would be produced by the electrometallurgical treatment of both driver and blanket sodium-bonded spent nuclear fuel.

Under the No Action Alternative, the EIS evaluates the potential impacts of the continued storage of spent nuclear fuel and high-level radioactive waste at the current storage locations using two scenarios: the first assumes continued storage under institutional controls for at least 10,000 years, and the second assumes no institutional controls after 100 years.

The SBSNF EIS considers the potential disposal at a geologic repository of spent nuclear fuel or high-level radioactive waste that may result from the proposed action involving sodium-bonded spent nuclear fuel. The Yucca Mountain Draft EIS includes the potential long-term impacts of repository disposal from electrometallurgically treated sodium-bonded spent nuclear fuel using data presented in the SBSNF Draft EIS. Quantities of radioactive waste analyzed in the Yucca Mountain document were based on previous projections that have been updated in this document.

1.6.2.3 Idaho High-Level Waste and Facilities Disposition Draft Environmental Impact Statement

This draft EIS was issued in December 1999 (DOE 1999d). It evaluates alternatives for managing the high-level radioactive waste and associated radioactive waste and facilities at INEEL. Under the terms of the 1995 Settlement Agreement and Consent Order with the State of Idaho, DOE agreed to treat high-level radioactive waste currently stored at INEEL and to prepare the waste in a form ready to be shipped out of the State of Idaho by 2035. The purpose of this EIS is to assist DOE in making decisions concerning the management of this radioactive waste to ensure compliance with applicable laws and regulations, and protect the environment and the health and safety of the workers and the public in a cost-effective manner. The high-level radioactive waste generated by the treatment and management of sodium-bonded spent nuclear fuel at ANL-W would not require any additional treatment at INEEL and is not evaluated in the Idaho High-Level Waste EIS.

In this EIS, DOE evaluates reasonable alternatives and options for the treatment of high-level radioactive waste, sodium-bearing waste, newly generated waste, and the disposition of facilities associated with high-level radioactive waste generation, treatment, and storage at INEEL. In addition, this EIS is integrated with the ongoing Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) program at the Idaho Nuclear Technology and Engineering Center. The proposed action under this EIS contributes to the cumulative impacts at the site discussed in the SBSNF EIS.

1.6.3 Electrometallurgical Treatment Research and Demonstration Project

Before electrometallurgical treatment could be considered as a technology choice for treating EBR-II spent nuclear fuel, an appropriate demonstration project was needed to evaluate its technical feasibility. As a preliminary step to demonstration, DOE requested that the National Academy of Sciences' National Research Council conduct an independent assessment of electrometallurgical treatment technology and its potential application to EBR-II spent nuclear fuel. In its report, published in 1995, the National Research Council recommended DOE proceed with demonstrating the technical feasibility of electrometallurgical treatment using a fraction of the EBR-II spent nuclear fuel. Following the National Research Council's recommendation, DOE conducted an environmental assessment of the demonstration project. The environmental assessment was

completed in May 1996 and resulted in a Finding of No Significant Impact, so that no further NEPA review was necessary for the demonstration project to proceed (Benedict et al. 1999).

In June 1996, DOE initiated a three-year testing program at ANL-W to demonstrate the technical feasibility of electrometallurgical treatment of up to 100 EBR-II driver spent nuclear fuel assemblies and up to 25 depleted uranium EBR-II blanket spent nuclear fuel assemblies. These two types of EBR-II spent nuclear fuel, driver and blanket, are typical of most of DOE's sodium-bonded spent nuclear fuel inventory (Benedict et al. 1999). The number of driver spent nuclear fuel assemblies was selected to provide the minimum fission product loading (approximately 3 percent) needed to evaluate the effectiveness of the removal of fission products from the electrorefiner salt and their concentration in the ceramic waste form. The blanket spent nuclear fuel assemblies were treated using a high-throughput electrorefiner that was installed in ANL-W's Fuel Conditioning Facility to evaluate higher-efficiency electrorefining (DOE 1996a).

A total of 100 driver spent nuclear fuel assemblies were treated. These assemblies required multiple batch operations of the treatment equipment in a remote, radioactive hot cell with an inert argon atmosphere. These operations were considered sufficient to demonstrate a dependable, predictable process, including uptime, repair and maintenance, and the operability of the linked process steps. A repeatability demonstration was completed by processing 12 driver spent nuclear fuel assemblies under the same processing conditions. In addition, processing 100 driver spent nuclear fuel assemblies dissolved sufficient active fission products in the electrorefiner salt so that ceramic waste form samples could be produced with representative waste loadings. The purpose of including blanket spent nuclear fuel assemblies in the test program was to demonstrate the mass throughput capacity of the process equipment and facility. A one-month throughput test was completed and a total of 13 blanket spent nuclear fuel assemblies were treated by the end of August 1999, when the demonstration project was concluded (Benedict et al. 1999).

To support the Electrometallurgical Treatment Research and Demonstration Project, DOE established an extensive research and development program at Argonne National Laboratory-East. The largest element of this research and development program involved development, testing, and qualification of the ceramic waste form. Another element was experimental support for electrorefining and metal processing operations in the Fuel Conditioning Facility. In addition, the research and development program included a modeling activity aimed at understanding and improving the electrometallurgical treatment process as well as laying out the requirements for production-scale treatment of the remaining EBR-II spent nuclear fuel. The combined results of the research and development program at Argonne National Laboratory-East and the spent fuel treatment operations at ANL-W provided the technical basis for final evaluation of the electrometallurgical treatment process. An extensive series of topical reports was prepared to present the results of the demonstration in detail. These reports were the basis for ANL-W's summary report on the demonstration project (Benedict et al. 1999).

To assist in monitoring the progress of the demonstration project, DOE requested that the National Research Council establish a review committee, the Committee on Electrometallurgical Techniques for DOE Spent Fuel Treatment, to evaluate the technology and its development. Working with DOE and the National Research Council committee, ANL-W established four criteria for evaluating the demonstration. The evaluation criteria for the electrometallurgical spent fuel demonstration project are listed below (Benedict et al. 1999).

Criterion 1: *Demonstrate that 100 driver and up to 25 blanket EBR-II assemblies can be treated in a Fuel Conditioning Facility within three years, with a throughput rate of 16 kilograms per month for driver assemblies sustained for a minimum of three months, and a blanket spent nuclear fuel throughput rate of 150 kilograms per month sustained for one month.*

Criterion 2: *Quantification (for both composition and mass) of recycle, waste, and product streams that demonstrate projected material balance with no significant deviations.*

Criterion 3: *Demonstrate an overall dependable and predictable process considering uptime, repair and maintenance, and operability of the linked process steps.*

Criterion 4: *Demonstrate that safety risks, environmental impacts, and nuclear materials accountancy are quantified and acceptable within regulatory limits.*

Based on a comparison of the demonstration results with the above criteria for success, the demonstration project was a technical success. All key performance criteria were met or exceeded. The results of the demonstration project proved the technical feasibility of using electrometallurgical treatment technology to process DOE's inventory of sodium-bonded spent nuclear fuel. In addition, the demonstration project validated the throughput rate of the sodium-bonded spent nuclear fuel, quantified all process streams, fine-tuned the operational parameters, refined the electrometallurgical treatment equipment, and provided actual waste forms for characterization. This last accomplishment was of particular importance because, as the Defense Waste Vitrification Project at SRS has shown, waste characterization is a lengthy process. Waste forms must be subjected to detailed chemical analysis and long periods of exposure to expected repository conditions. The waste form characterization in the electrometallurgical treatment demonstration project has already initiated the waste acceptance process. Preliminary results of waste form testing indicate that both the metal and ceramic waste forms produced by the electrometallurgical process appear to be comparable to borosilicate glass, which has been tested and analyzed extensively under conditions relevant to a geologic repository.

The review committee of the National Academy of Sciences' National Research Council has continuously reviewed the progress of the Electrometallurgical Treatment Research and Demonstration Project and all reports to date have found the process to be proven for treating sodium-bonded spent nuclear fuel (Benedict et al. 1999).

In the most recent status report issued in the summer of 1999 (NAS 1999), the National Research Council Committee expressed some concerns about the long-term performance and potential releases from the waste forms under repository conditions. However, as noted above, work completed at ANL-W since the latest National Research Council review of the project indicates that both the ceramic and metallic electrometallurgical treatment waste forms would be comparable to borosilicate glass, which has been tested extensively under conditions relevant to the repository. The final report from the National Research Council Committee was published in April 2000. The National Research Council's final report on *Electrometallurgical Techniques for DOE Spent Fuel Treatment* concluded that "The EBR-II demonstration project has shown that the electrometallurgical technique can be used to treat sodium-bonded spent nuclear fuel." The report further stated that "The committee has found no significant technical barriers in the use of electrometallurgical technology to treat EBR-II spent fuel, and EMT [electrometallurgical treatment] therefore represents a potentially viable technology for DOE spent nuclear fuel treatment." DOE will consider the Council's final report during the Record of Decision process which follows the issuance of the final EIS.

1.7 CHANGES FROM THE DRAFT EIS

In response to comments on the SBSNF Draft EIS and as a result of information that was unavailable at the time of the draft EIS issuance, the final EIS contains revisions and new information. These revisions and new information are indicated by a double underline for minor word changes or by a sidebar in the margin for sentence or larger additions. Appendix A contains the comments received during the public review period of the SBSNF Draft EIS and DOE's responses to these comments. Responses to comments related to cost and nuclear nonproliferation issues, although included in the Appendix, did not result in any changes to the EIS. A brief discussion of the most important changes included in the final EIS is provided in the following paragraphs.

Results of the Electrometallurgical Treatment Research and Demonstration Project

As a result of public concern that results of the demonstration project were not incorporated in the draft EIS, a section (Section 1.6.3) was added in the final EIS with a description, status, and results of the demonstration project.

Justification of Purpose and Need and Timing

As a result of public concern that the draft EIS did not adequately justify the need and timing for the proposed action, Section 1.2 was revised to reflect DOE's position and DOE's responses to the related comments.

Relationship to Other NEPA Actions

The final EIS was revised to update the information provided on the *Draft Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada,* which was issued in July 1999 (DOE 1999b).

Sodium Removal and Disposition

As a result of public comment, the description of an alternate method for decladding and cleaning sodium-bonded blanket spent nuclear fuel, the laser declad and alcohol wash process, was added in Section 2.3.9. The reason the process was not included in the evaluation of the reasonable alternatives also is included in Section 2.3.9.

No Action Alternative Definition

One of the two options of the No Action Alternative was revised from "indefinite" storage until the development of a currently less mature technology to "continued storage of the sodium-bonded spent nuclear fuel until 2035 or until the development of a currently less mature technology." The revision clarifies the issue raised by public comments concerning the time period covered by this EIS. This EIS covers the time period until 2035.

In addition, under both options of the No Action Alternative, it was determined that the sodium-bonded spent nuclear fuel would be packaged at ANL-W in preparation for shipment out of the State of Idaho by 2035.

No Action Alternative Assumptions

As a result of public comment, the assumption for the calculation of the radiological gaseous emissions under the No Action Alternative was changed. The draft EIS conservatively assumed that the radiological gaseous emissions would be a fraction of the total radiological gaseous emissions presented in the Programmatic Spent Nuclear Fuel EIS, in direct proportion to the heavy metal mass ratio of the sodium-bonded spent nuclear fuel to the total spent nuclear fuel stored at INEEL. The final EIS directly calculates the radiological gaseous emissions using a more realistic fuel degradation assumption based on historical evidence. This change considerably reduced the estimated radiological gaseous emissions as well as the resulting doses to workers and the public under the No Action Alternative.

Dose and Risk Calculations

As a result of public comments and the availability of recent data from the Electrometallurgical Treatment Research and Demonstration Project and the *Savannah River Site Spent Nuclear Fuel Management Final Environmental Impact Statement* (DOE 2000), dose calculations were revised in the final EIS. These revisions

include: (1) the addition of project total doses to workers, (2) the project total risk to the public under normal operations, and (3) changes in doses and risks to the public and workers from accidents. In addition, dose and risk values were rounded, resulting in some changes in the numerical values in the EIS.

Air Quality

Based on public comments on the draft EIS, concentrations and emissions from sources that operate in support of the processing alternatives at ANL-W (e.g., emergency generators) were quantified and added to Sections 3.2.3 and 3.3.3 (Air Quality and Noise) and Chapter 4 of the final EIS. In addition, the baseline nonradiological air quality concentrations for INEEL presented in the draft EIS were replaced with more current emission inventory data.

Land Use/Ecology

As a result of comments received on the draft EIS, reference to the newly established 29,950-hectare (74,000-acre) INEEL Sagebrush Steppe Ecosystem Reserve was added to Sections 3.2.1.1 (Land Use) and 3.2.6 (Ecological Resources) of the final EIS.

Water Quality

As a result of public comments, a discussion and a summary table of radioactive liquid effluent at both INEEL and SRS were added to Sections 3.2.4 and 3.3.4 (Water Resources) of the final EIS.

Geology and Soils

As a result of public comments on the draft EIS, material on earthquake activity and volcanism in the vicinity of INEEL (Section 3.2.5, Geology and Soils) was revised.

Existing Human Health Risk

As a result of public comments, baseline concentrations and associated hazard indexes or cancer risks for hazardous chemicals at both ANL-W and SRS were added to Sections 3.2.10 and 3.3.10 (Existing Human Health Risk) of the final EIS.

Waste Management

Records of Decision for the Final Waste Management Programmatic EIS (DOE 1997) addressing the management of high-level radioactive waste and low-level and mixed low-level radioactive waste were issued on August 26, 1999 (64 FR 46661), and February 25, 2000 (65 FR 10061), respectively. A summary of these decisions was added to the waste management discussion for both INEEL and SRS (Sections 3.2.11 and 3.3.11, respectively).

Cumulative Impacts

The cumulative impacts section (Section 4.11) was updated to reflect recent information obtained from the *Idaho High-Level Waste and Facilities Disposition Draft Environmental Impact Statement* (DOE 1999d) and the *Savannah River Site Spent Nuclear Fuel Management Final Environmental Impact Statement* (DOE 2000).

Electric Energy Consumption

Section 4.14.3, Irreversible and Irretrievable Commitments of Resources, was revised to include electrical energy consumption associated with the alternatives under the proposed action.

Settlement Agreement and Consent Order with the State of Idaho

As a result of public comments concerning the contents of the Settlement Agreement and Consent Order with the State of Idaho, the entire document was included in the final EIS as Appendix K.

Melt and Dilute Driver Fuel at SRS

The option of using the melt and dilute process to treat sodium-bonded driver spent nuclear fuel at SRS was considered at the recommendation of a public comment. The option was dismissed from further evaluation as explained in the revised Section 2.6 of the final EIS.

Preferred Alternative

In accordance with requirements of the Council on Environmental Quality regulations (40 CFR 1502.14e), the final EIS incorporates DOE's Preferred Alternative for the treatment and management of sodium-bonded spent nuclear fuel. The Preferred Alternative is discussed in Section 2.8.

Transportation

The analysis was expanded to include the impacts from transporting the various waste forms and spent nuclear fuel packages from ANL-W to the INEEL Dry Storage Facility prior to transporting materials out of the State of Idaho by 2035.

Miscellaneous Revisions and Editorial Changes

Several sections in the SBSNF Final EIS were revised to reflect the availability of more recent data or to include corrections, improvements in the presentation, and other editorial changes. None of these revisions affects the environmental analysis presented in the EIS.

1.8 ORGANIZATION OF THE EIS

This EIS volume contains 9 chapters and 12 appendices. The main analyses are included in the chapters and additional project information is provided in the appendices. The 9 chapters provide the following information:

Chapter 1—Introduction: Background on the disposition of spent nuclear fuel; purpose and need for the proposed action; issues identified during the scoping and public comment periods; decisions to be made; and relationship of this EIS to other DOE NEPA actions and programs

Chapter 2—Proposed Action and Alternatives: Descriptions of sodium-bonded spent nuclear fuel; spent nuclear fuel treatment methods; spent nuclear fuel management facilities; alternatives considered; background information on the ultimate disposition of spent nuclear fuel; Preferred Alternative; and summary comparison of environmental impacts

Chapter 3—Affected Environment: Aspects of the environment that could be affected by the EIS alternatives

Chapter 4—Environmental Consequences: Analyses of the potential impacts of the EIS alternatives on the environment

Chapter 5—Environmental Laws, Regulations, and Consultations: Environmental, safety, and health regulations that would apply for this EIS's alternatives and the agencies consulted for their expertise

Chapters 6-9—Glossary; a list of preparers; a list of agencies, organizations, and persons to whom copies of this EIS were sent; and an index

The 12 appendices contain the following information: overview of the public participation process (scoping meetings and public comment period) and comment disposition; methods for assessing environmental impacts; detailed technology descriptions; characteristics of sodium-bonded spent nuclear fuel; normal operational impacts on human health; facility accident impacts on human health; evaluation of human health effects of overland transportation; environmental justice analysis; scientific terminology for ecological resources; *Federal Register* notices; the Settlement Agreement and Consent Order with the State of Idaho; and the contractor disclosure statement.

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Chapter 2

Proposed Action and Alternatives

2. PROPOSED ACTION AND ALTERNATIVES

Chapter 2 describes the proposed action and alternatives to treat and manage the U.S. Department of Energy's inventory of sodium-bonded spent nuclear fuel and the No Action Alternative. The chapter includes discussions on the characteristics and quantities of the sodium-bonded spent nuclear fuel under consideration, the proposed treatment methods, and the potential sites and facilities for treatment or storage. It discusses why certain alternatives were dismissed from consideration. It also addresses issues associated with the ultimate disposition of the spent nuclear fuel and provides a summary comparison of the environmental impacts associated with the proposed action and the No Action Alternative.

2.1 INTRODUCTION

To fulfill the purpose and need discussed in Section 1.2, the U.S. Department of Energy (DOE) proposes to treat and manage sodium-bonded spent nuclear fuel and facilitate ultimate disposal in a geologic repository. The reasonable alternatives are determined by the technology options available to DOE to treat and manage the sodium-bonded spent nuclear fuel. To assist the reader in understanding the proposed action and reasonable alternatives, the following sections provide background information on the characteristics, inventory, and current storage locations of the sodium-bonded spent nuclear fuel; a discussion on the Electrometallurgical Treatment Research and Demonstration Project; the technology alternatives for its treatment and management; and the locations where these technologies could be implemented. The reasonable alternatives are discussed in Section 2.5.

2.2 SODIUM-BONDED SPENT NUCLEAR FUEL

As a result of research, development, and demonstration activities associated with liquid metal fast breeder reactors, DOE has approximately 60 metric tons of heavy metal of sodium-bonded spent nuclear fuel in its inventory. This represents approximately 2 percent of DOE's total current spent nuclear fuel inventory of nearly 2,500 metric tons of heavy metal. The common characteristic of sodium-bonded spent nuclear fuel is the presence of metallic sodium in the space between the cladding and the fuel and/or within the mass of the fuel. The presence of this chemically reactive material necessitates DOE's consideration of suitable treatment and management alternatives for this spent nuclear fuel before disposal in a geologic repository. Detailed descriptions of the characteristics of the sodium-bonded spent nuclear fuel in DOE's inventory are included in Appendix D.

Proposed Action

DOE proposes to treat and manage the sodium-bonded spent nuclear fuel in a safe and efficient manner and facilitate disposal in a geologic repository.

The bulk of sodium-bonded spent nuclear fuel in DOE's inventory is of two general types: driver fuel and blanket fuel. Driver fuel is used mainly in the center of the reactor core to "drive" and sustain the fission chain reaction. It is highly enriched in the fissile isotope uranium-235. Blanket fuel is made from depleted uranium, a type of uranium in which most of the fissile uranium-235 has been removed. Blanket fuel usually is placed at the perimeter of the core and is used to breed the fissile material plutonium-239. It contains primarily the nonfissile isotope uranium-238, which converts to fissile plutonium-239 with the absorption of neutrons produced from the fission process. In some cases, as in the case of the Experimental Breeder Reactor-II (EBR-II), blanket fuel also has been used at the perimeter of the core for shielding. Typically, the fuel matrix in the sodium-bonded spent nuclear fuel is a uranium alloy or uranium

metal. A very small quantity (approximately 0.1 percent in mass of heavy metal) is in the form of uranium oxide, uranium or plutonium nitride, and uranium or plutonium carbide. Typical driver and blanket spent nuclear fuel elements are shown schematically in **Figure 2–1.**

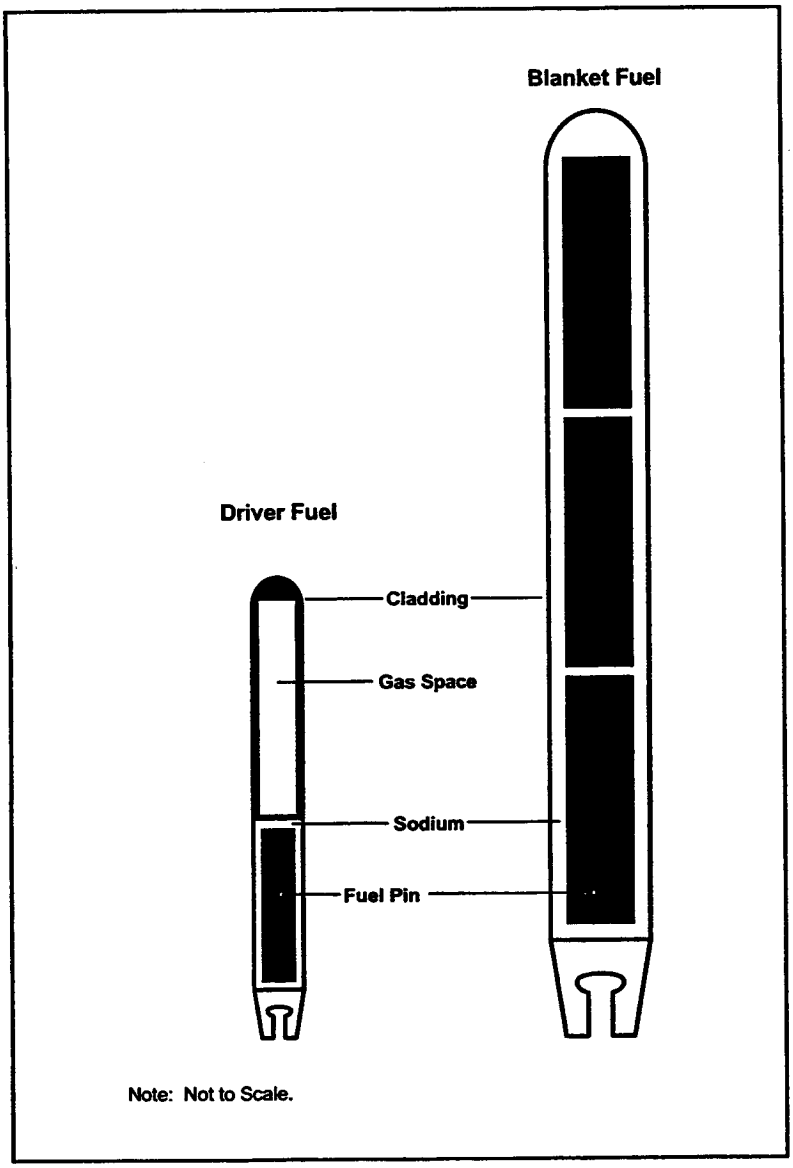

Figure 2–1 Typical Driver and Blanket Spent Nuclear Fuel Elements

The blanket and driver spent nuclear fuel addressed in this environmental impact statement (EIS) contain metallic sodium between the cladding and the metallic fuel pins to improve the heat transfer from the fuel to the reactor coolant through the stainless steel cladding. When driver fuel is irradiated in the reactor for some period of time, the metallic fuel swells as fission products are generated until it reaches the cladding wall. Pores form throughout the fuel as it swells under pressure from the gaseous fission products. As these pores expand and connect to one another, the fission gases escape to a plenum in the fuel element just above the metallic fuel. As the gas escapes, the liquid sodium flows into these tiny pores, much like a sponge. As more pores form and grow, others are closed off from the fuel surface, including those containing sodium. Between 20 and 40 percent of the available sodium (up to 0.8 grams) may enter the driver fuel and become inseparable from the uranium except by dissolving or melting the fuel (Hofman and Walters 1994).

It also is well documented (Hofman and Walters 1994) that fuel and cladding components interdiffuse during irradiation to such an extent that mechanical stripping of the driver fuel cladding is not practical. The gap between the cladding and the fuel pin that contains sodium early in the irradiation lifetime disappears gradually due to outward swelling of the fuel pin. After approximately a 1 to 3 percent burnup, this gap is closed by swelling of the fuel pin due to irradiation effects such as fission gas bubble growth. Once contact between the fuel pin outer surface and cladding inner surface is made, cladding constituents (mainly nickel, chromium, and iron) gradually interdiffuse with fuel constituents (mainly uranium, plutonium, and zirconium) and the rare earth fission products (neodymium, cerium, lanthanum, praseodymium, samarium, and promethium) in the fuel. A solid-state layer bonding the fuel and cladding together is formed. This interdiffused layer effectively attaches the cladding to the fuel pin permanently in localized regions. Mechanical forces applied to these regions in a decladding operation either would leave pieces of the fuel pin attached to the cladding or vice versa. The resulting mix of cladding and fuel still would contain sodium inaccessible to subsequent treatment.

Blanket fuel, on the other hand, is at such a low burnup that significant swelling of the fuel pin does not occur. Gaps between the fuel pin and the cladding still exist at low burnup and little or no interdiffusion takes place. Cladding therefore can be mechanically removed from blanket fuel. Also, the swelling of this fuel is still low enough at typical blanket burnup that very little interconnected porosity exists. Hence, minimal sodium trapping would take place. Therefore, blanket fuel could be declad to effect sodium removal. The sodium removed from the sodium-bonded blanket spent nuclear fuel could be distilled and stabilized for disposal as low-level radioactive waste. The cladding would be disposed of as low-level radioactive waste or as part of the waste form being qualified for the proposed geologic repository.

2.2.1 EBR-II Spent Nuclear Fuel

EBR-II was a research and test reactor at Argonne National Laboratory-West (ANL-W) used to demonstrate the engineering feasibility of a sodium-cooled, liquid metal reactor with a steam electric power plant and integral fuel cycle. It achieved initial criticality in September 1961 and continued to operate until September 1994. During the 33 years of operation, numerous fuel designs were tested in EBR-II. EBR-II spent nuclear fuel contains both driver and blanket fuel.

The EBR-II driver spent nuclear fuel is stainless steel-clad, highly enriched uranium in a uranium alloy, typically either zirconium or fissium (an alloy of molybdenum, ruthenium, rhodium, palladium, zirconium, and niobium). There are some variations in the specific cladding alloys, enrichments, fuel compound alloys, dimensions, and burnup. When the fuel is "spent," the enrichment (ratio of uranium-235 to total uranium) ranges between 55 and 76 percent. Each driver spent nuclear fuel element has a metal fuel pin about 36 centimeters (14 inches) long and less than 0.5 centimeters (0.2 inches) in diameter. The typical EBR-II driver spent nuclear fuel pin is a metal alloy of either 90 percent uranium and 10 percent zirconium or 95 percent uranium and 5 percent fissium. This fuel pin and a small amount of metallic sodium are loaded into a 74-centimeter-long (29-inch-long) stainless steel tube (cladding) and welded shut. This unit of fuel is called an element. Sixty-one (91 for some fuel) fuel elements are put together in a stainless steel hexagonal duct to make a fuel assembly approximately 2.3 meters (92 inches) long and 5.8 centimeters (2.3 inches) across. The principal isotopes contributing to the activity of the axial and radial blanket assemblies are given in Appendix D.

The EBR-II blanket spent nuclear fuel consists of stainless steel-clad, depleted uranium in metallic form. There are various blanket designs: upper and lower axial, and inner and outer radial blankets. The primary differences between the blanket designs are the dimensions. In EBR-II, the blanket assemblies were used primarily for shielding and for reducing the required size of the reactor core. Blanket assemblies were placed outside of a stainless steel shield for all but the first few years of EBR-II operation. Blanket assemblies are similar to driver assemblies, except that the individual blanket pins are larger. The blanket pins, made entirely from depleted uranium, are 1.1 centimeters (0.4 inches) in diameter. Three to five pins placed end-to-end make a sodium-bonded blanket element between 84 and 140 centimeters (33 to 55 inches) long. Since the blanket pins have a larger diameter, 19 blanket elements comprise a blanket assembly. The principal isotopes contributing to the activity of the axial and radial blanket assemblies are given in Appendix D.

The fuel from the last seven years of EBR-II operation is presently stored in three different locations at ANL-W (the Fuel Conditioning Facility, the Hot Fuel Examination Facility, and the Radioactive Scrap and Waste Facility) and two different locations at the Idaho Nuclear Technology and Engineering Center (INTEC), formerly the Idaho Chemical Processing Plant (CPP). Previously, spent nuclear fuel was shipped to INTEC for reprocessing. However, INTEC ceased accepting the fuel in 1991 when a new uranium-zirconium alloy fuel, which could not be dissolved with INTEC's existing plutonium-uranium extraction (PUREX) reprocessing system, went into full use at EBR-II. Prior to that, approximately 6 metric tons of EBR-II fuel were processed at INTEC. When DOE stopped processing at INTEC in 1992, elements from some 500 EBR-II driver spent nuclear fuel assemblies of earlier design were left in storage pools (CPP-603 and

CPP-666) located at INTEC. Water has been observed leaking into some of the storage containers in the CPP-603 storage pool, and the EBR-II fuel inside has reacted with the water and produced hydrogen gas. This is one of the reasons DOE is planning to remove all the spent nuclear fuel from the CPP-603 storage pool and place it in dry storage. National Environmental Policy Act (NEPA) coverage for this activity is provided by 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 Spent Nuclear Fuel EIS) (DOE 1995a), and is not within the scope of this EIS.

2.2.2 Fermi-1 Spent Nuclear Fuel

Fermi-1 was designed and built at Monroe, Michigan (30 miles southwest of Detroit), to demonstrate the feasibility of the sodium-cooled, liquid metal fast breeder reactor for electric power production. Fermi-1 was a sodium-cooled, fast reactor. The reactor achieved initial criticality in 1963 and operated until September 1972. Fermi-1 was licensed for operation at a power level of 200 megawatts-thermal. Only blanket fuel from Fermi-1 is sodium-bonded.

The Fermi-1 blanket spent nuclear fuel consists of stainless steel-clad, depleted uranium in a uranium-molybdenum alloy. There are various blanket designs: upper and lower axial, and inner and outer radial blankets. The primary differences between these blanket designs are dimensions, elements per assembly, and burnup. Fermi-1 blanket elements are similar to EBR-II blanket elements in enrichment, but differ in dimensions (Fermi-1 elements are larger), form (uranium-molybdenum alloy versus uranium metal), and burnup. Because of its lower burnup, the Fermi-1 blanket fuel, which contains only about 0.2 percent plutonium by weight compared to approximately 1 percent plutonium by weight for the EBR-II blanket fuel, is subject to less stringent safeguard and security requirements than the EBR-II blanket fuel. This is an important consideration in the cost of managing the storage of these two types of fuel.

After the Fermi-1 reactor was permanently shut down, the blanket assemblies were placed into 14 canisters and transported to INTEC in 1974 and 1975 in 14 shipments. The canisters are made of stainless steel with a carbon steel basket inside. The canisters are 3.46 meters (134 inches) long and 64.8 centimeters (25.5 inches) in diameter. Twelve of the canisters contain the radial blanket subassemblies and 2 of the canisters contain the shorter axial blanket subassemblies. A subassembly is a cut assembly containing the blanket fuel pins. The canisters were placed into CPP-749, which is an underground dry storage system. The 14 canisters are stored in a single row of vaults on 4.6-meter (15-foot) centers.

The total quantity of Fermi-1 blanket material, both axial and radial, is 34 metric tons of heavy metal. The blanket assemblies have a very low irradiation history. Therefore, the inventory of fission products, activation products, and transuranics is low. The principal isotopes contributing to the activity of the axial and radial blanket assemblies are given in Appendix D.

2.2.3 Fast Flux Test Facility and Miscellaneous Sodium-Bonded Spent Nuclear Fuel

DOE's inventory of sodium-bonded spent nuclear fuel includes eight liquid metal reactor test assemblies containing driver spent nuclear fuel that were irradiated at the Fast Flux Test Facility at Hanford, Washington. It also includes small quantities of fuel that have metallic sodium or the alloy of sodium and potassium from liquid metal reactor experiments. These miscellaneous small-lot fuels differ in cladding composition, uranium content, enrichment, and burnup. Some of the fuel consists of uranium and/or plutonium carbides, nitrides, and oxides in addition to metal uranium or alloy. For the purposes of this EIS, this miscellaneous fuel is assumed to have driver fuel characteristics. This fuel is located at several DOE sites such as the Hanford site, Oak Ridge National Laboratory, the Savannah River Site (SRS), Sandia National Laboratories/New Mexico, and the Idaho National Engineering and Environmental Laboratory (INEEL). Those lots stored outside INEEL will be transported to INEEL pursuant to the amended Record of Decision (61 FR 9441) for the Programmatic

Spent Nuclear Fuel EIS (DOE 1995a). Under the proposed action, it is assumed that this spent fuel will be stored at INEEL.

Table 2–1 provides a summary of the spent nuclear fuel addressed by this EIS. As described earlier, the majority of the spent nuclear fuel consists of EBR-II driver fuel, EBR-II blanket fuel, and Fermi-1 blanket fuel.

Table 2–1 Overview of Sodium-Bonded Spent Nuclear Fuel Categories

Spent Nuclear Fuel Type	*Storage Volume[a] (cubic meters)*	*Metric Tons of Heavy Metal*	*Sodium Content (kilograms)*
EBR-II driver	58[b]	3	83
EBR-II blanket	13	22	173
Fermi-1 blanket	19	34	365
Fast Flux Test Facility driver	8[b]	0.3	7
Miscellaneous[c]	3[b]	0.1	31
Total	101	60	662

[a] Volume refers to the canister storage volume.
[b] A larger volume per unit mass for the driver spent nuclear fuel is required for criticality control.
[c] Assumed to have driver fuel characteristics.

Table 2–2 provides the site where the sodium-bonded spent nuclear fuel is stored, the locations within the DOE site, and the various storage configurations within the storage site.

Table 2–2 Sodium-Bonded Spent Nuclear Fuel Storage Locations and Configurations

Spent Nuclear Fuel Type	*Current Storage Locations and Configurations*		
	DOE Site	*Location*	*Configuration*
EBR-II driver	INEEL (ANL-W)	Radioactive Scrap and Waste Facility	Loose elements in canisters
		Hot Fuel Examination Facility	Loose elements
		Fuel Conditioning Facility	In process material[a]
EBR-II blanket	INEEL (ANL-W)	Radioactive Scrap and Waste Facility	Elements in canisters
		Fuel Conditioning Facility	In process material[a]
EBR-II driver	INEEL (INTEC)	CPP-603 basin	About 12 elements per canister
		CPP-666 basin	
Fermi-1 blanket	INEEL (INTEC)	CPP-749 dry well underground	Cut/uncut assemblies in 14 storage canisters
Fast Flux Test Facility driver	INEEL (ANL-W)	Hot Fuel Examination Facility	Loose elements
	Hanford	Fast Flux Test Facility, Buildings 405 and 403	Intact assemblies
Miscellaneous	Sandia National Laboratories/ New Mexico	Technical Area V	Experimental capsule
	SRS	Receiving Basin for Offsite Fuel	Elements
	Oak Ridge National Laboratory	Building 3525	Elements

[a] Processed as part of the EBR-II Electrometallurgical Treatment Research and Demonstration Project.

2.3 TREATMENT AND MANAGEMENT METHODS

DOE has identified several potential treatment, management, and packaging methods that could be used to prepare sodium-bonded spent nuclear fuel for disposal in a geologic repository. These are: the electrometallurgical process; the PUREX process; packaging in high-integrity cans; the melt and dilute process; the glass material oxidation and dissolution system (GMODS) process; the direct plasma arc-vitreous ceramic process; and the chloride volatility process. Each of these methods is discussed below. In formulating reasonable alternatives under the proposed action, the GMODS process, the direct plasma arc process, and the chloride volatility process were not considered sufficiently mature technologies to be included as reasonable alternatives (see Section 2.6).

Direct disposal of sodium-bonded spent nuclear fuel in a geologic repository without treatment, i.e., packaging the fuel in high-integrity cans without sodium removal, has been considered in this EIS under the No Action Alternative. The option may not meet current U.S. Nuclear Regulatory Commission (NRC) and/or Resource Conservation and Recovery Act (RCRA) requirements.

2.3.1 Electrometallurgical Treatment Process

The electrometallurgical treatment process was developed at the Argonne National Laboratory for processing EBR-II driver and blanket spent nuclear fuel assemblies with metallic fuel. The process has been demonstrated for the stainless steel-clad uranium alloy fuel used in that reactor. Modifications to the process are used for the treatment of oxide, nitride, and carbide sodium-bonded spent nuclear fuel. The electrometallurgical treatment process uses electrorefining, which is an industrial technology used to produce pure metals from impure metal feedstock (DOE 1996). Electrorefining has been used to purify metal for more than 100 years. **Figure 2–2** illustrates the various steps within the electrometallurgical treatment process at ANL-W.

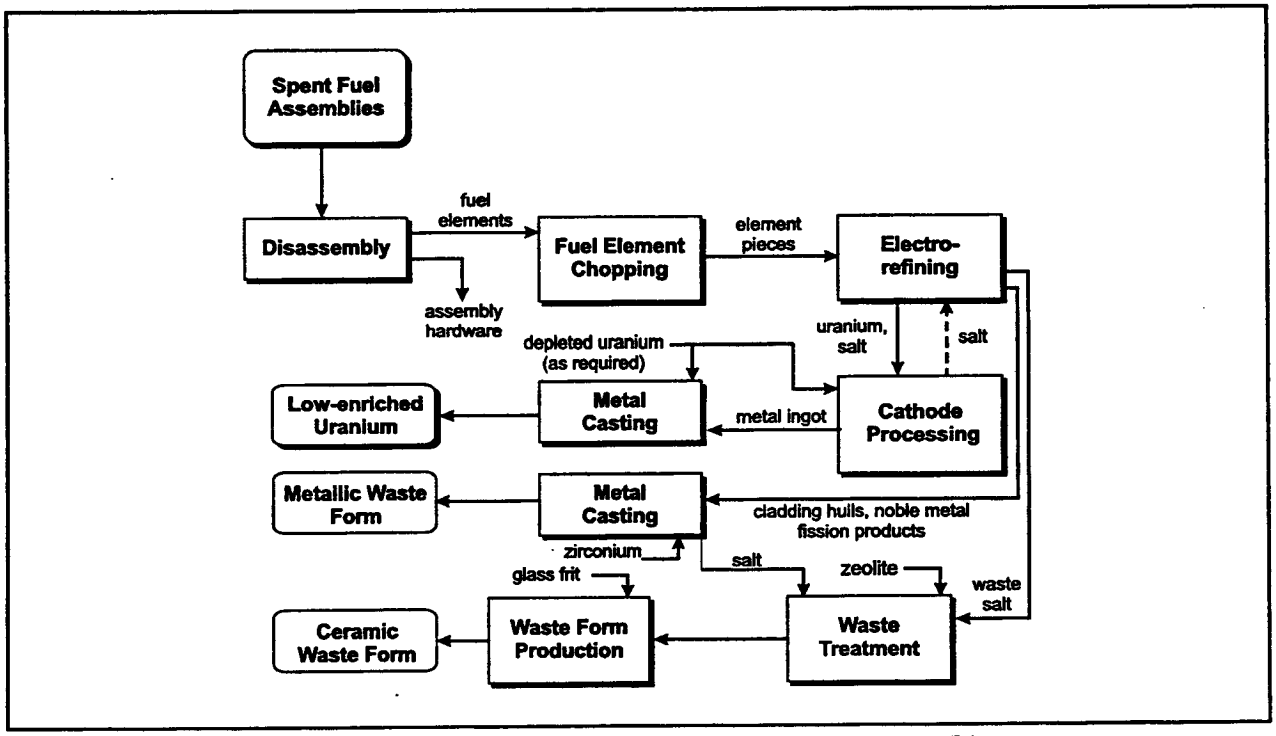

Figure 2–2 Electrometallurgical Treatment Process Flow Diagram

The first step in processing sodium-bonded spent nuclear metallic fuel involves the removal of fuel elements from the fuel assemblies. The fuel elements then would be chopped into short segments and placed in stainless steel baskets to form the anode in the electrorefiner.

The electrorefiner, in which the electrometallurgical treatment occurs, would be maintained at about 500 °C (930 °F) and contain a molten mixture of primarily two salts, lithium chloride and potassium chloride. The chopped fuel elements in the anode baskets would be lowered into the molten salt. Upon application of an electric voltage between the anodes and cathodes, uranium, transuranic elements including plutonium, most of the fission products, and the sodium would dissolve into the salt. The uranium would be deposited by the current at the cathode. The stainless steel cladding hulls and some of the insoluble fission products (i.e., noble metals) would remain in the anode baskets.

After a sufficient amount of spent nuclear fuel has been treated, the salt would be removed and solidified. The salt, which contains the sodium in the form of sodium chloride, transuranic elements, and most fission products extracted from the spent nuclear fuel, would be solidified, ground to a desired size, and mixed with zeolite. Zeolite is any of a group of alumina silicate minerals used as filters and ion-exchange agents. Zeolite is used to collect certain fission products from the process salt. The fission products, sodium, and transuranics, including plutonium in the salt and zeolite, would be heated so the salt becomes sorbed into the zeolite structure. Glass frit then would be added to the zeolite mixture and hot-pressed to produce a ceramic high-level radioactive waste form that is expected to be suitable for disposal.

The uranium deposited at the cathode would be removed from the electrorefiner and treated to remove any adhered salts. Then the uranium would be melted (and depleted uranium added if necessary), solidified to form an ingot, and further processed in a metal casting furnace to produce low-enriched uranium ingots. The stainless steel cladding hulls and the insoluble fission products would be melted in the casting furnace to produce a metal high-level radioactive waste form that is expected to be suitable for disposal in a geologic repository.

The oxide fuel would be prepared for treatment using the electrometallurgical treatment process by reducing the uranium oxide to uranium metal with lithium metal dissolved in small batches of lithium chloride-potassium chloride molten salt solution. The resulting uranium-bearing solution would be added to the molten salt solution used in the electrometallurgical treatment process for other sodium-bonded fuel and blanket elements and would be processed with those materials.

The carbide fuel would be prepared for electrometallurgical treatment by cleaning the fuel of sodium to the extent possible and then converting the fuel to uranium oxide with water or diluted acid. This oxide then would be converted to uranium metal by lithium metal in a molten salt solution and processed by electrometallurgical treatment with other sodium-bonded spent nuclear fuel and blanket elements. The nitride fuel also would be prepared for electrometallurgical treatment by converting it to uranium fuel.

In addition to the metal and ceramic waste form, some low-level radioactive waste also would be generated during the disassembly process of the spent nuclear fuel assemblies in the form of hardware. A detailed description of the electrometallurgical treatment process is presented in Appendix C.

2.3.2 PUREX Process

The PUREX process is a counter-current solvent extraction method which has been used extensively throughout the world since 1954 to separate and purify uranium and plutonium from fission products contained in aluminum-clad spent nuclear fuel and irradiated uranium targets. PUREX is not a thermal process; therefore, it takes place at low temperatures. DOE has two operating facilities at SRS, F-Canyon and H-Canyon, that use the PUREX process for treatment of aluminum-clad fuel and targets. Use of the PUREX

process facilities at SRS for treating sodium-bonded spent nuclear fuel involves certain restrictions inherent in the design: 1) the presence of sodium complicates the process as employed in the front-end of the SRS facilities; 2) the presence of stainless steel cladding would require significant modifications or additions to the existing front-end of the facilities; and (3) the presence of alloys (e.g., zirconium) in some of the fuel is incompatible with the SRS dissolution process. For this reason, treatment of driver sodium-bonded spent nuclear fuel is not feasible without significant modification to the existing PUREX process. However, the SRS facilities could be used without modification for the blanket sodium-bonded spent nuclear fuel if it is declad and its sodium removed prior to the process. In such a case, the F-Canyon facility would be used.

The fuel pins would be dissolved in an aqueous solution of nitric acid. The resulting nitric acid solution containing uranium, plutonium, and fission products would undergo feed clarification (to remove settlable solids) and acidity/alkalinity adjustment. The clarified aqueous solution then would be treated via the PUREX process utilizing centrifugal contactors and separators that involve organic solvent washing to produce: (1) an aqueous high-level radioactive waste containing the bulk of the fission products, americium, and neptunium; (2) a material stream containing the recovered plutonium; and (3) a material stream containing the recovered uranium. The plutonium- and uranium-containing streams each would undergo a second cycle of solvent washing to further separate the residual fission products and actinides from the plutonium and uranium. The aqueous high-level radioactive waste eventually would be processed to a borosilicate glass form. Material streams from the PUREX process would be uranium oxide, plutonium metal, and high-level radioactive waste. The uranium oxide would be stored on site as depleted uranium. The plutonium, approximately 260 kilograms (572 pounds), would be disposed of in accordance with the Record of Decision (75 FR 1608) for the *Surplus Plutonium Disposition Final Environmental Impact Statement* (DOE 1999c). **Figure 2–3** illustrates the various steps necessary for the treatment of sodium-bonded spent nuclear fuel in conjunction with the PUREX process. A detailed description of the process is presented in Appendix C.

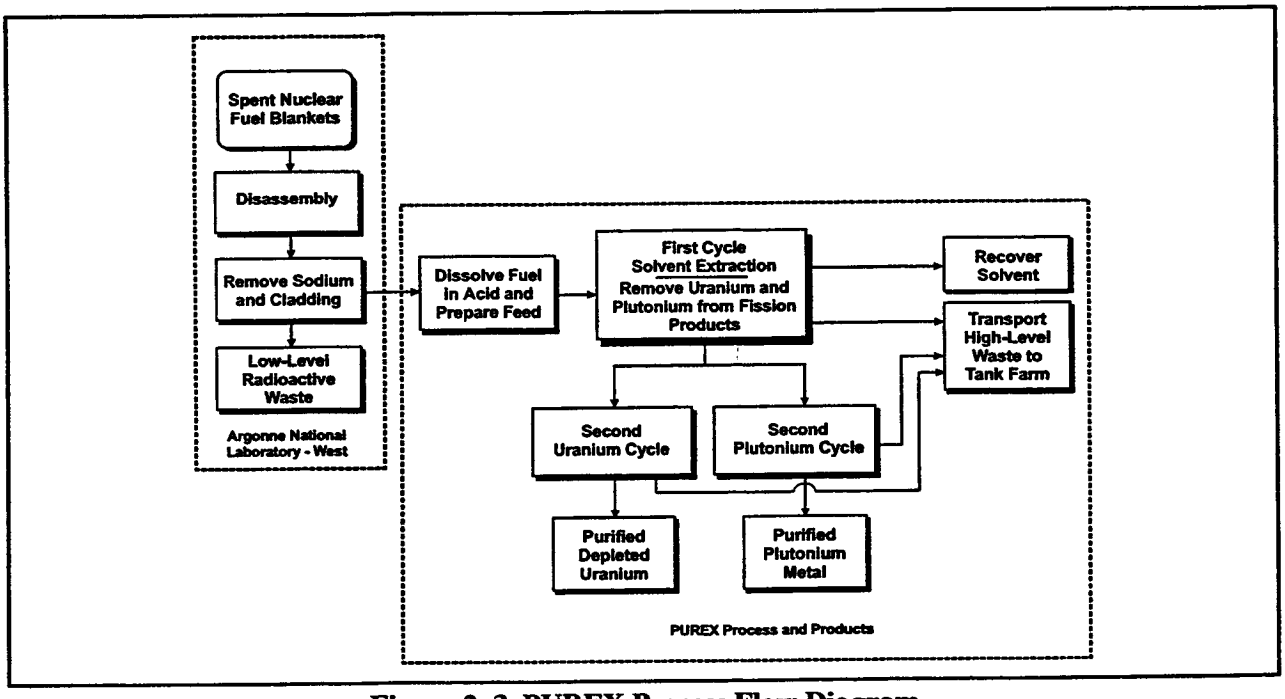

Figure 2–3 PUREX Process Flow Diagram

2.3.3 High-Integrity Can Packaging

The high-integrity can packaging provides substitute cladding for damaged or declad fuel, or another level of containment for intact fuel. The can would be used to store fuel on site until it is ready to be shipped to the repository. The can is constructed of a highly corrosion-resistant material (Hastelloy Alloy C-22) to provide corrosion protection during storage at the site. It also could provide long-term protection in a repository environment (i.e., for 1,000 or more years after repository closure with no institutional control). The high-integrity cans would be placed into a standardized canister for transportation and eventual placement in the repository in waste packages.

The analysis for packaging sodium-bonded spent nuclear fuel in high-integrity cans was performed with and without decladding and/or sodium removal. Packaging sodium-bonded blanket spent nuclear fuel in high-integrity cans with sodium removal was analyzed in the EIS under Alternative 2. Packaging sodium-bonded spent nuclear fuel in high-integrity cans without sodium removal was analyzed in this EIS as a direct disposal option under the No Action Alternative (see Sections 2.3.8 and 2.5.1).

The high-integrity cans would be placed in dry storage at ANL-W. Prior to emplacement in a repository, the high-integrity cans would be placed into a standardized canister designed to promote containment under repository conditions. **Figure 2–4** illustrates the high-integrity can flow process. A detailed description of the high-integrity can packaging is presented in Appendix C.

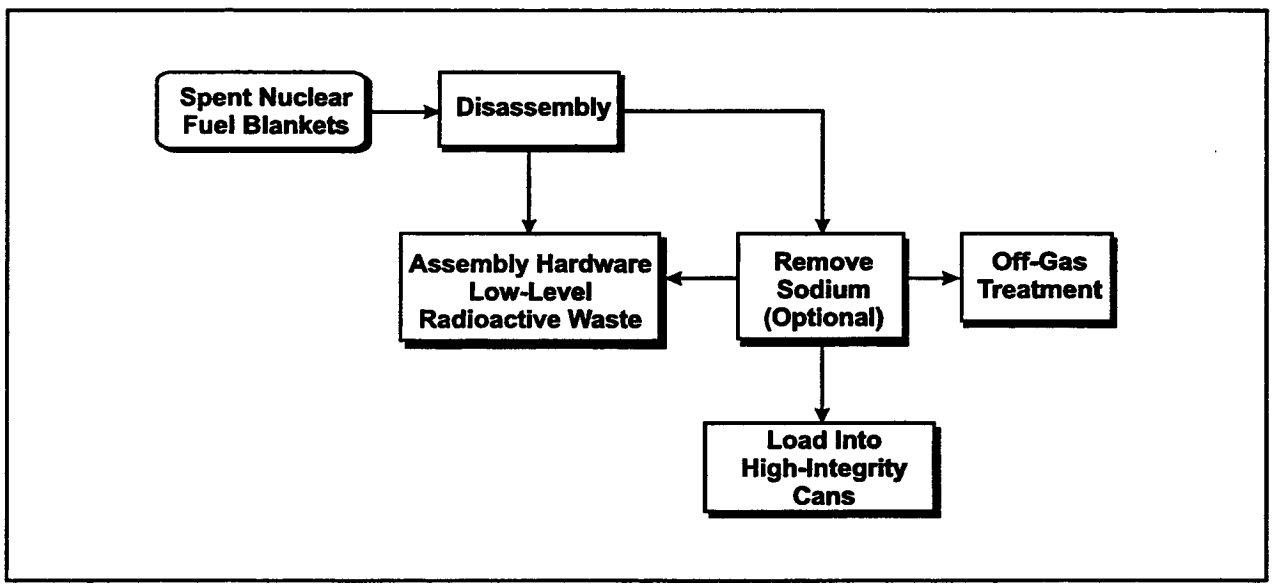

Figure 2–4 High-Integrity Can Packaging Flow Diagram

2.3.4 Melt and Dilute Process

There are three options for the melt and dilute process. In the first option, bare uranium blanket spent nuclear fuel pins with the sodium removed would be melted with aluminum at SRS using technology similar to that proposed for the aluminum-clad research reactor fuel. The second and third options would be conducted at the ANL-W site using metallurgical technology developed for uranium and stainless steel cladding. In the second option, blanket spent nuclear fuel elements would be melted with additional stainless steel. In the first two options, there would be no actual dilution of the fissile component of the uranium since it is present at only 0.2 percent, i.e., far less than the 0.7 percent in natural uranium. **Figure 2-5** illustrates the first two options of the melt and dilute process. The third option would involve a modified melt and dilute process that would

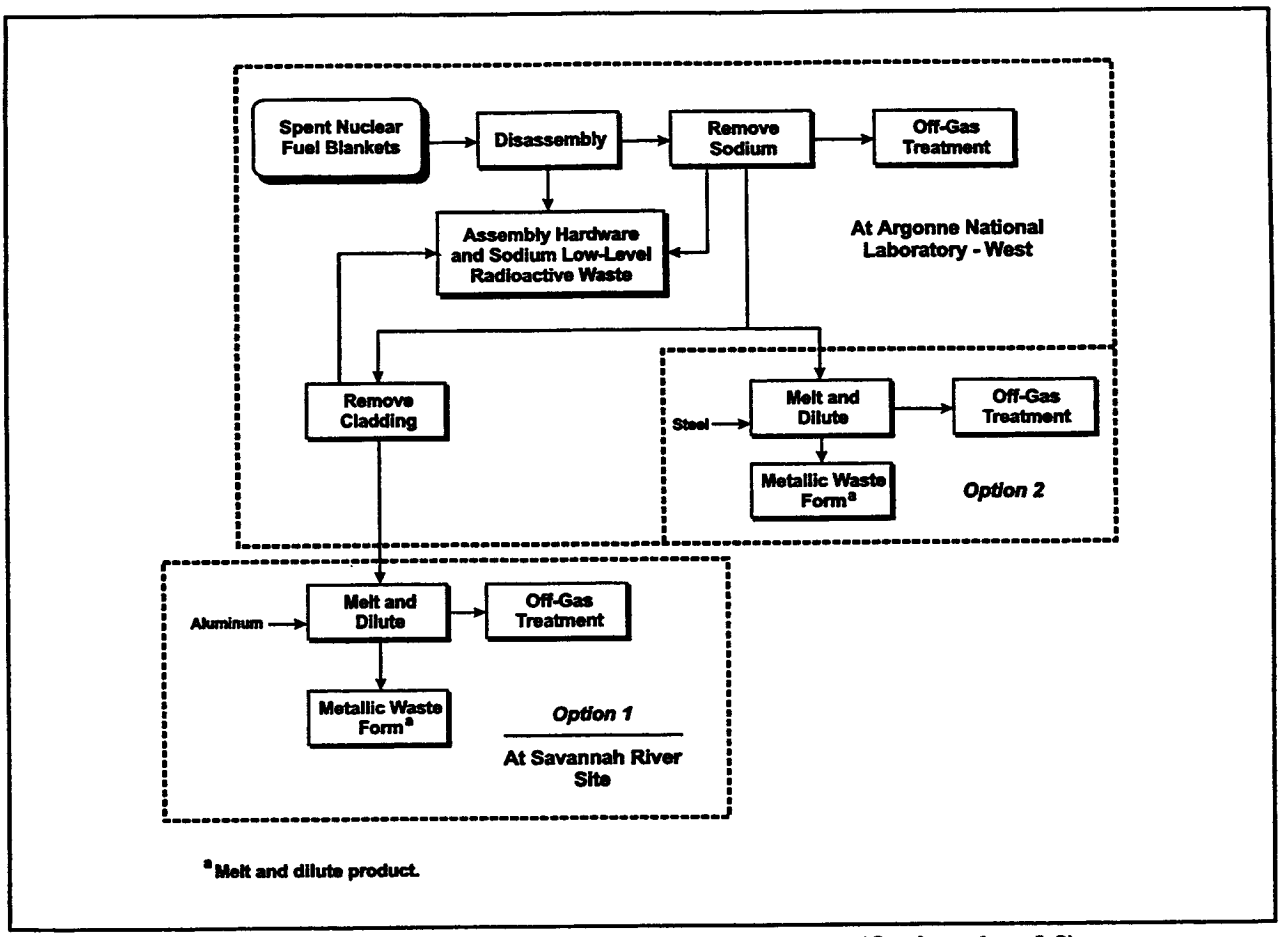

Figure 2–5 Melt and Dilute Process Flow Diagram (Options 1 and 2)

be capable of handling the sodium volatilized from processing chopped driver spent nuclear fuel elements with the cladding intact. **Figure 2-6** illustrates the third option of the melt and dilute process. A detailed description of the melt and dilute process is presented in Appendix C.

Option 1: Uranium-Aluminum Option for Blanket Pins

Declad and cleaned blanket pins would be received at SRS in aluminum canisters, each containing some 60 kilograms (132 pounds) of material. The canisters would be stored until they fit into the processing schedule. Following validation of the contents, the canisters would be loaded into a melting furnace with additional aluminum. The furnace would operate at a temperature of approximately 1,000 °C (1,830 °F), significantly in excess of the aluminum-uranium alloy melting temperature, to initiate melting within a reasonable time frame. Volatile fission products would be captured by a series of filter banks before releasing the off-gas. A metal alloy ingot would be cast, sampled, and packaged.

Option 2: Uranium-Steel Option for Blanket Pins

Blanket elements with the sodium removed but not declad would be loaded into a furnace crucible. A small amount of radioactive waste steel might be added to the crucible to reach the desired uranium-steel composition. The furnace would be heated to approximately 1,400 °C (2,550 °F) to melt the uranium, after which the steel would be dissolved slowly into the uranium pool. The mixture would be stirred electromagnetically to a uniform composition. Volatile fission products would be captured by a series of filter banks before releasing the off-gas. A metal alloy ingot would be cast, sampled, and packaged.

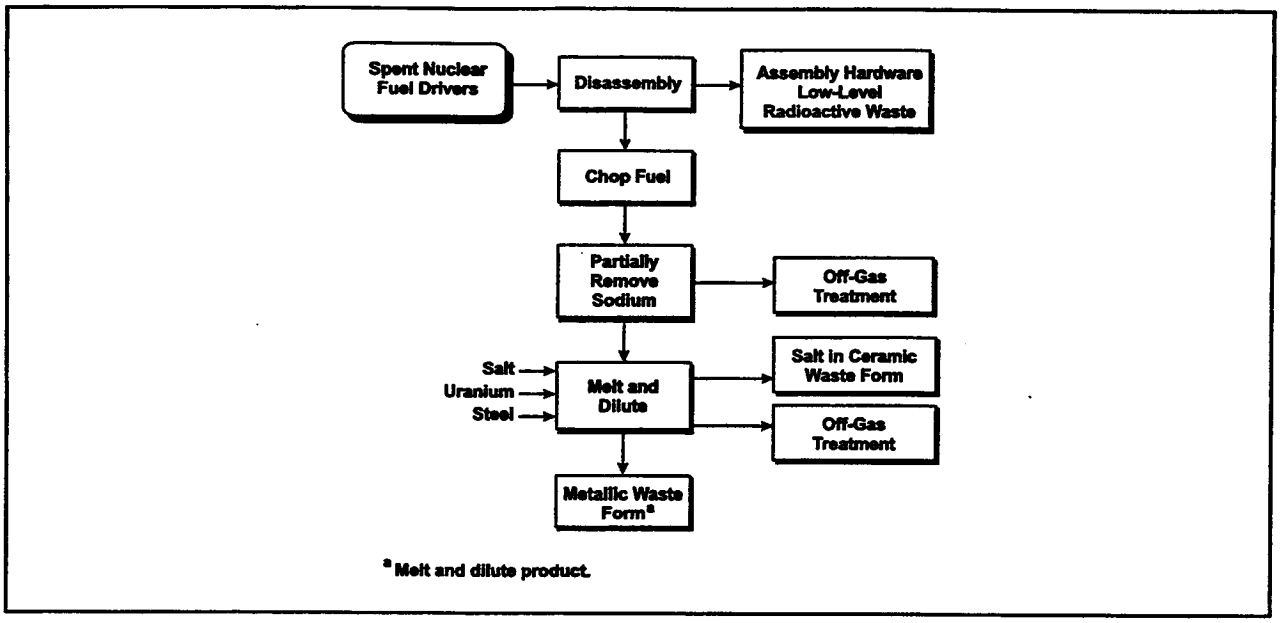

Figure 2–6 Melt and Dilute Process Flow Diagram (Option 3)

Option 3: Uranium-Steel Option for Driver Spent Nuclear Fuel

Some of the sodium in driver spent nuclear fuel elements would be removed in a similar manner to the sodium from blanket spent nuclear fuel elements. A modified melt and dilute process would be used for driver spent nuclear fuel still containing the cladding and some metallic sodium. The addition of flux salt is the only modification to the process required to capture residual sodium from the driver spent nuclear fuel. Chopped driver spent nuclear fuel elements would be loaded into an induction furnace and covered with a layer of low melting-temperature salt containing uranium, iron, or manganese chloride as a component to oxidize the molten sodium. Depleted uranium would be added in the ratio of about 2.5 to 1 to reduce the enrichment to less than 20 percent uranium-235. Radioactive waste steel would be added to complete the mixture. The use of radioactive waste steel reduces the inventory of the low-level radioactive waste. This furnace is operated at the same temperature as in Option 2. The sodium would react with and be captured in the flux salt, protecting the off-gas treatment filter banks. After the melt is mixed, a vacuum would be applied to complete the volatilization of the salt, which would be condensed and partially reused. The salt, which includes sodium in a nonreactive form, would be stabilized in a ceramic waste form similar to the waste form from the electrometallurgical treatment process. The metal melt would be stirred to achieve uniform composition and cast into an ingot, placed in a standardized canister, and stored.

The process described above can be used for sodium-bonded spent nuclear metallic fuel. Oxide, carbide, and uranium nitride fuel types cannot be treated using the melt and dilute process because of their high melting points.

2.3.5 GMODS Process

The GMODS process uses oxides to convert unprocessed spent nuclear fuel directly to borosilicate glass. The basic concept is to combine unprocessed sodium-bonded spent nuclear fuel and lead borate glass in a glass melter at a temperature of 800 to 1,000 °C (1,470 to 1,830 °F). The uranium and plutonium in the spent nuclear fuel would be converted into oxides and dissolved in the glass. Due to the powerful dissolution and oxidation properties of the lead borate glass melt, containment is a concern, and a water-cooled, cold-wall, induction-heated melter must be used. The waste form is borosilicate glass and would contain uranium, the transuranic elements, the fission products, and the sodium present in the sodium-bonded spent nuclear fuel.

As with all processes that dissolve or melt spent nuclear fuel, the GMODS process would produce radioactive off-gases. These gases would be filtered and treated by appropriate means. **Figure 2–7** illustrates the GMODS treatment process. A description of this process is presented in Appendix C.

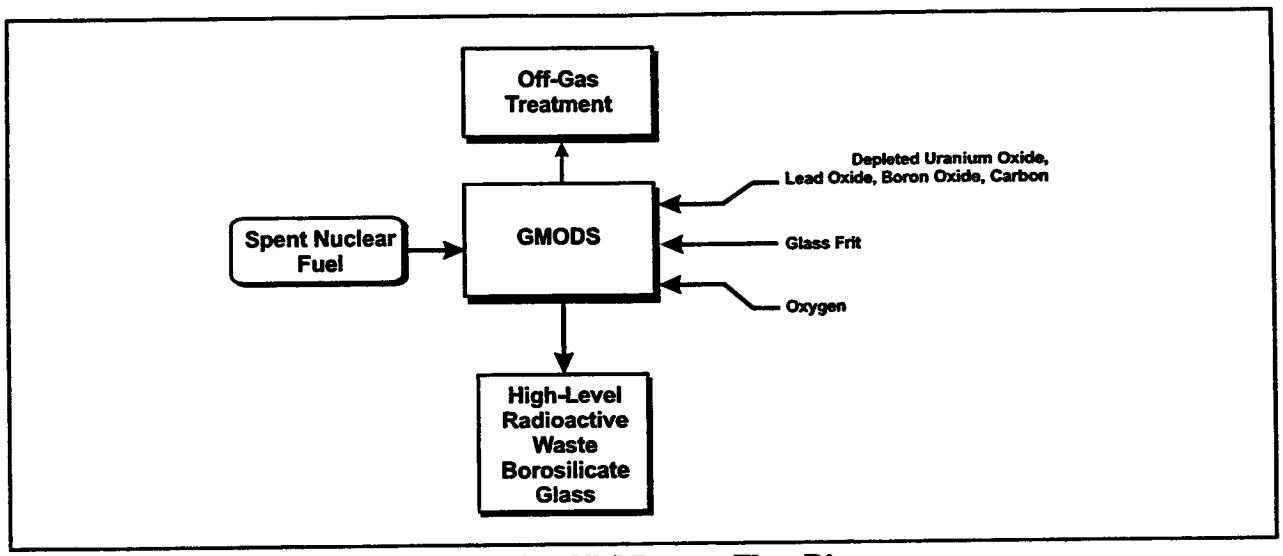

Figure 2–7 GMODS Process Flow Diagram

2.3.6 Direct Plasma Arc-Vitreous Ceramic Process

In this process, the sodium-bonded spent nuclear fuel would be cut into small pieces and melted and oxidized in a rotating furnace containing molten ceramic materials at a temperature of about 1,600 °C (2,900 °F) or higher. A direct-current plasma torch would supply the energy required in the process. Rotation would be used to keep the molten pool in the furnace. The spent nuclear fuel would be fed into the process with minimal pretreatment. Ceramic material would be added as necessary with the mixture homogenized by the torch. When the spent nuclear fuel is homogeneously melted and oxidized throughout the ceramic, rotation would be slowed to allow the molten vitreous ceramic to pour out by gravity flow into canister molds.

Metallic fuel such as the EBR-II spent nuclear fuel would require the addition of some ceramic material. Depleted uranium could be added to the process in almost any form to reduce the uranium-235 enrichment if necessary. Criticality issues would be addressed by limiting the process to batch runs of preselected quantities of fissile material with the addition of the depleted uranium and neutron poisons, if necessary.

As with all processes that dissolve or melt spent nuclear fuel, the plasma arc treatment would produce radioactive off-gases. These gases would be filtered and treated by appropriate means. **Figure 2–8** illustrates the direct plasma arc-vitreous ceramic treatment process. A description of this process is presented in Appendix C.

2.3.7 Chloride Volatility Process

The chloride volatility process is an advanced treatment technology that was investigated at INEEL. The process uses the differences in volatilities of chloride compounds to segregate major nonradiological constituents from spent nuclear fuel for the purpose of volume reduction, and isolates the fissile material to produce a glass or ceramic waste form. The major steps are: (1) high-temperature chlorination at about 1,500 °C (2,730 °F) and conversion of metallic fuel and cladding to gaseous chloride compounds; (2) removal of the transuranic chlorides and most of the fission products in a molten zinc chloride bed at approximately 400 °C (750 °F); (3) condensation of the other chlorides (e.g., uranium hexachloride) in a series of fluidized

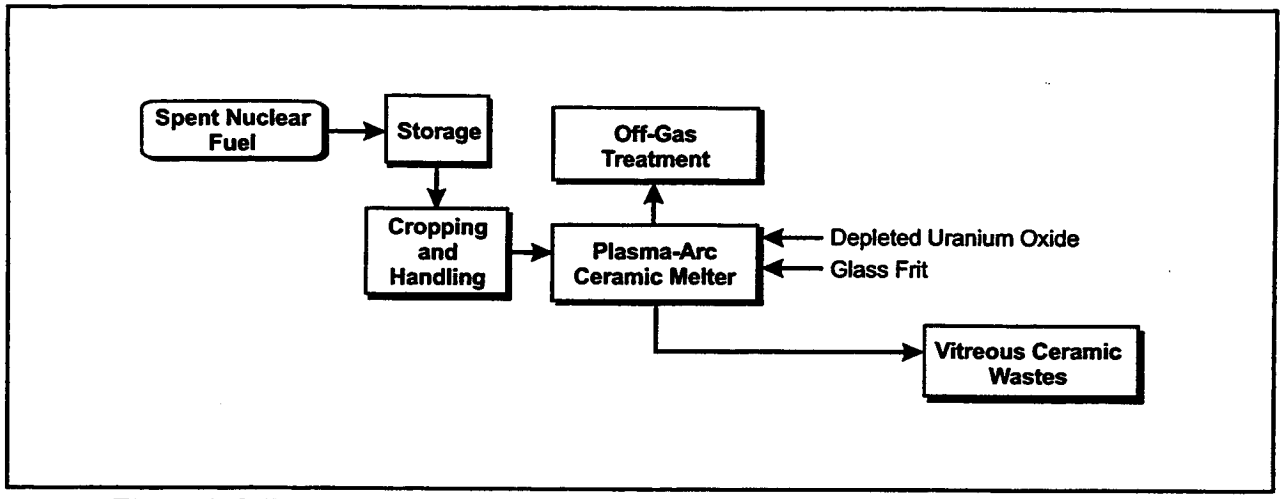

Figure 2–8 Direct Plasma Arc-Vitreous Ceramic Treatment Process Flow Diagram

beds and condensers at successively lower temperatures; and (4) zinc chloride regeneration/recycling. The transuranics and fission product chlorides then would be converted into either fluorides or oxides for disposal. This process inherently handles volatilized fission products and chlorine gas, which presents significant unique occupational and public risks. **Figure 2–9** illustrates the chloride volatility treatment process. A description of this process is presented in Appendix C.

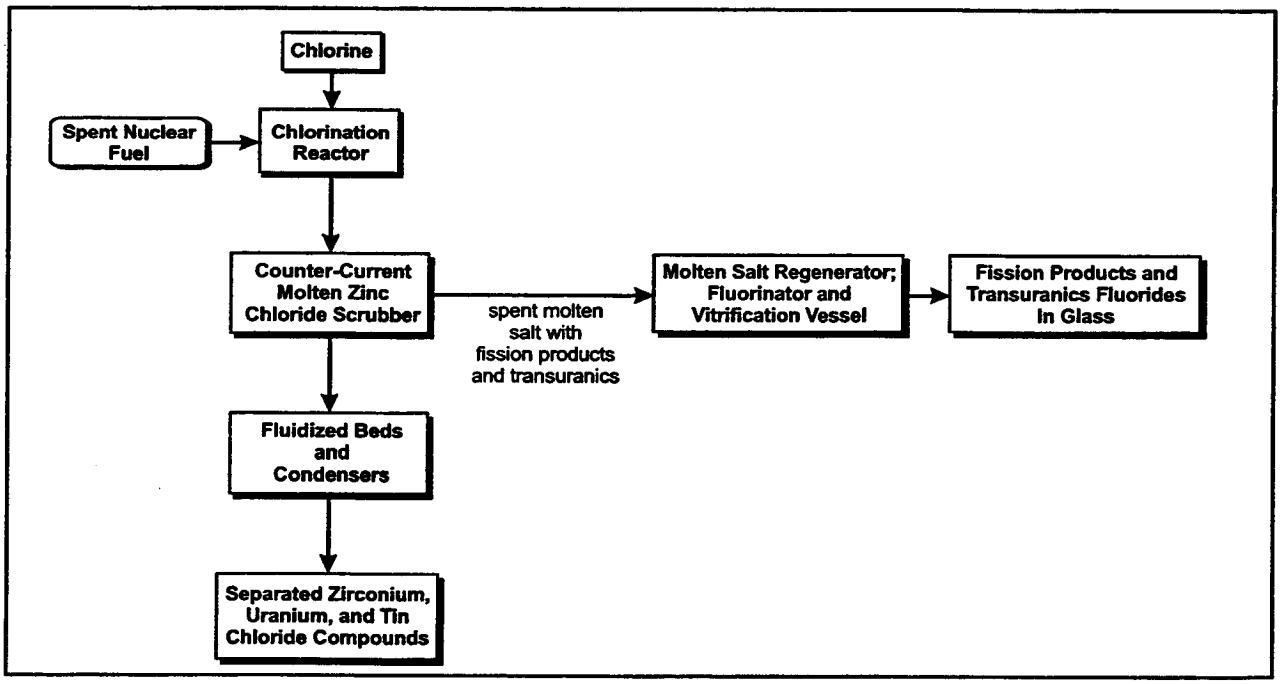

Figure 2–9 Chloride Volatility Process Flow Diagram

2.3.8 Direct Disposal

For the purpose of this EIS, direct disposal of sodium-bonded spent nuclear fuel is disposal without sodium removal. The sodium-bonded spent nuclear fuel (driver and blanket) would be packaged in high-integrity cans as described in Section 2.3.3 without removing the metallic sodium. The high-integrity cans would be placed into a standardized canister designed to provide containment under repository conditions during preclosure operations. At the present time, RCRA mixed waste (which contains both hazardous and radioactive waste) does not meet the requirements of acceptable waste as identified in the current April 1999 *Civilian Radioactive Waste Management System-Waste Acceptance System Requirements Document,* (DOE 1999a). Because of the presence of metallic sodium, the sodium-bonded spent nuclear fuel could be categorized as a RCRA hazardous waste that is potentially both pyrophoric and reactive (DOE 1997). Additionally, the NRC prohibits the disposal of materials that contain or generate explosive, pyrophoric, or chemically reactive substances that could compromise the repository's performance. Therefore, direct disposal would not meet current DOE or NRC repository acceptance criteria.

2.3.9 Sodium Removal and Disposition

As discussed in Section 2.2 and the preceding sections, the disposition of the metallic sodium in the sodium-bonded spent nuclear fuel varies with the treatment method. For those methods that do not require the removal of metallic sodium prior to treatment, or decladding of the fuel (e.g., the electrometallurgical process), the metallic sodium would be converted into a nonreactive salt as part of the process and would be incorporated in the high-level radioactive waste product of the process.

For the methods that require the removal of sodium prior to treatment and/or decladding of the spent fuel (i.e., the PUREX process, the melt and dilute process for blanket spent nuclear fuel [Options 1 and 2], and the packaging in high-integrity cans) the removed metallic sodium would be processed separately, converted into a nonreactive salt, and disposed of as low-level radioactive waste or high-level radioactive waste along with the waste form associated with the treatment process. Decladding and sodium removal could be done using either a mechanical process (the melt, drain, evaporate, and calcinate [MEDEC] process) or a laser declad and alcohol wash process.

In the MEDEC process, the blanket fuel is brought into an argon-atmosphere hot cell where the ends of the cladding for each fuel element are cut off to expose the sodium within the cladding. An argon-atmosphere is required for work involving materials such as sodium which could react with the moisture in air. Then the fuel is placed into a crucible furnace where it is subjected to a temperature of about 200 °C (390 °F), causing melting of the sodium, which is drained into a collection tank. After this bulk sodium is removed, the fuel temperature is raised to about 500 °C (930 °F) and a 10^{-4} torr vacuum is applied to the chamber housing this fuel. This higher temperature vacuum step volatilizes residual sodium and removes it from the fuel. This vacuumed sodium vapor would be condensed in a trap and collected with the previously removed bulk sodium pending further processing.

Sodium recovered during the cleaning process may contain some fission products, most notably cesium-137. The sodium would be stabilized using an oxidation/carbonation process (ANL 1999). Under this process, the cesium would be recovered by vacuum distillation of the sodium, taking advantage of the large difference in the boiling points of the two elements. The boiling point of cesium is 690 °C (1,274 °F), while the boiling point of sodium is 892 °C (1,638 °F). A vapor trap would be placed between the distillation column and pump to collect volatile species emitted from the condenser. The purified sodium would be processed by injection into a chamber, where it would react rapidly with oxygen and water to form aqueous sodium hydroxide. Carbon dioxide gas then would be bubbled through the hydroxide solution converting the sodium hydroxide to sodium carbonate. The aqueous sodium carbonate would be solidified with a binder and

packaged for disposal as low-level radioactive waste. The cesium fraction collected as distillate from the separation process would be added to the high-level radioactive waste form from the process.

To remove the cladding after sodium has been extracted, a special machine would be installed in the argon cell. This machine would mechanically push the fuel pins within the cladding out through the opening created when the cladding ends of the fuel elements previously were cut off. Experience with unirradiated blanket fuel at Argonne National Laboratory has shown that the pins could be mechanically pushed out of the stainless steel cladding after the sodium has been eliminated.

For the melt and dilute process for driver spent nuclear fuel (Option 3), the sodium removed prior to the process would be processed separately, converted into a nonreactive salt, and disposed of as low-level radioactive waste. Any sodium remaining within the fuel would be removed during the melt and dilute process as nonreactive salt, stabilized in a ceramic waste form, and disposed of as high-level radioactive waste.

The laser declad and alcohol wash process has been performed at Rockwell International Hot Laboratory in Canoga Park, California. The activities in this process would be similar to those that were performed at Rockwell International. The process would use a modified laser system for remote operation and a cutting machine to hold and index the fuel elements. The fuel elements would be brought into a hot cell and cut in a predefined cutting sequence. The fumes generated during the cutting process would be filtered and exhausted through an off-gas system. The fuel pins along with the cladding strips would be washed in an alcohol/water mixture to neutralize the metallic sodium and fission product (i.e., cesium) contamination. The fuel pins would be packaged and stored, or sent to SRS for treatment. The alcohol/water solution would be partially evaporated, and the sodium/cesium alcoholates and hydroxides would be neutralized, then solidified in a grouting agent, and disposed of as low- or high-level radioactive waste, depending on the cesium content.

Several aspects of the Rockwell laser process would not meet current environmental standards and would violate the design requirements of an argon hot cell. First, the Rockwell laser process required personnel entry into the hot cell on a biweekly basis for laser maintenance and purging of the cell atmosphere to maintain a low oxygen level (less than 4 percent) and to vent alcohol/water vapors and hydrogen gas from the cell. Neither of these practices would be acceptable for argon cell operation today, in part because of stricter radiation exposure controls and a higher concentration of fission products in the remaining inventory of EBR-II blanket fuel relative to the fuel that was treated by Rockwell. Operation of an argon cell requires maintenance of a low moisture and low oxygen content atmosphere as well as limitations on liquids within the cell for criticality control. The alcohol wash process introduces a liquid which is evaporated into the cell. Second, sodium collected during previous laser decladding operations was able to be disposed of as low-level radioactive waste. The sodium collected from processing the fuel addressed by this EIS would be contaminated with cesium. If sufficient quantities of cesium were present in the sodium, this waste could not be managed as low-level radioactive waste. For the sodium to be managed as low-level radioactive waste, the sodium would have to be processed (as is done with the sodium removed from the fuel in the MEDEC process) using a currently undefined process to remove the cesium from the alcohol mixture. While criticality concerns related to high moisture content levels within a multipurpose argon cell could be eliminated by removing any stored fissile materials, frequent purging of the hot cell atmosphere and personnel entry would still be restricted by current radiation exposure controls and the high concentration of fission products involved. Only the MEDEC process was used to evaluate the various alternatives that require cleaning and/or decladding of the sodium-bonded spent nuclear fuel because of compatibility concerns about laser operation in the cell.

Table 2–3 a summarizes sodium removal and disposition for the treatment methods addressed in this EIS.

Table 2–3 Sodium Removal and Disposition by Treatment and Management Method

Treatment and Management Methods	*Decladding*	*Sodium Treatment*	*Sodium Disposition*
Electrometallurgical process Driver and blanket fuel	No	Stabilization	Converted into nonreactive form, as part of the process, and disposed of with the high-level radioactive ceramic waste product of the process.
High-integrity cans Blanket fuel	No	Removal	Converted into nonreactive form, separately from the process, and disposed of as low-level radioactive waste.
PUREX process Blanket fuel	Yes	Removal	Converted into nonreactive form, separately from the process, and disposed of as low-level radioactive waste.
Melt and dilute process Driver fuel	No	Removal	Part of the sodium is converted into nonreactive form, separately from the process, and disposed of as low-level radioactive waste. The remaining sodium is separated during the process, converted to nonreactive ceramic waste form, and disposed of as high-level radioactive waste.
Blanket fuel	Yes [a]/No [b]	Removal	Converted into nonreactive form, separately from the process, and disposed of as low-level radioactive waste.
Direct disposal [c] Driver and blanket fuel	No	No	Disposed of in metallic reactive form in high-integrity cans.

[a] Melt and dilute process at SRS.
[b] Melt and dilute process at ANL-W.
[c] The direct disposal option may not meet current NRC and/or RCRA requirements.

2.4 Spent Nuclear Fuel Treatment and Management Facilities

For each alternative, DOE would use existing spent nuclear fuel management facilities that provide remote-handling and heavy-lifting capability, hot cells, and space to receive sodium-bonded spent nuclear fuel shipments. These facilities would prepare, treat, and/or place the sodium-bonded spent nuclear fuel in interim storage awaiting treatment as needed. Besides treating the sodium-bonded spent nuclear fuel, these facilities would provide capabilities to open the shipping containers, sample and analyze the fuel, and vacuum-dry the spent nuclear fuel. These facilities also could be used to repackage the fuel into storage canisters and place the repackaged fuel in dry interim storage to await treatment. The spent nuclear fuel management facilities described in the following sections provide the capability to implement the proposed action for each of the previously described technology alternatives.

2.4.1 ANL-W

The ANL-W site is a center of nuclear technology development and testing (**Figure 2–10**). Five nuclear test reactors have operated on the site, although the only one currently active is a small reactor used for radiography examination of experiments, waste containers, and spent nuclear fuel. Work on highly radioactive materials is conducted in the Fuel Conditioning Facility and the Hot Fuel Examination Facility, both heavily shielded hot cell facilities. Inventories of nuclear materials are maintained on site for conducting research, as well as for storage, pending decisions for further disposition.

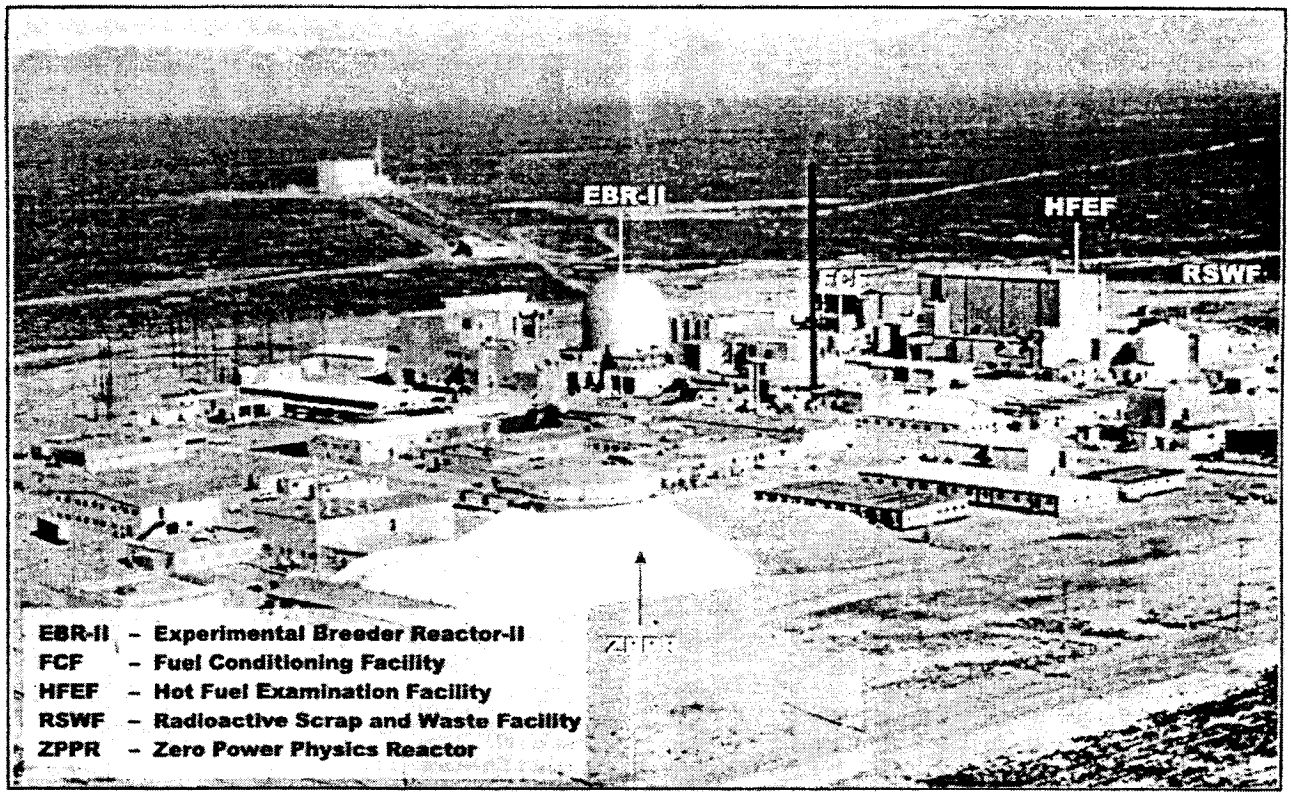

Figure 2–10 ANL-W

2.4.1.1 Fuel Conditioning Facility

The Fuel Conditioning Facility is one of the proposed facilities for the treatment and management of sodium-bonded spent nuclear fuel. The Fuel Conditioning Facility was activated in 1963 and consists of two hot cells, one with an air atmosphere and the other with an inert argon gas atmosphere. Since 1990, the Fuel Conditioning Facility has undergone major reconstruction and refurbishment to meet current safety and environmental requirements. A photographic view of the Fuel Conditioning Facility is shown in **Figure 2–11.** The hot cells enable technicians to work safely with radioactive nuclear materials from behind 1.5-meter-thick (5-foot-thick) shielding walls. A schematic plan of the main floor of the Fuel Conditioning Facility is shown in **Figure 2–12.** The rectangular air cell is used for handling, storage, and assembly/disassembly of components. The argon cell is a much larger hot cell and is "doughnut"-shaped; that is, personnel can work from the outside corridor around the hot cell and can monitor the work in the hot cell from an inner shielded work space in the center of the hot cell.

All equipment in the cells can either be repaired remotely using externally controlled robotic arms (manipulators) and cranes or can be removed and decontaminated for repair elsewhere in the facility from outside corridors around the hot cells.

In addition, the facility contains a mockup area where new equipment can be qualified and tested for remote operation and maintenance prior to installation in the cells. There is also a spray chamber, special glove boxes, and a suited-entry repair area (located in the basement) where contaminated equipment can be decontaminated and repaired.

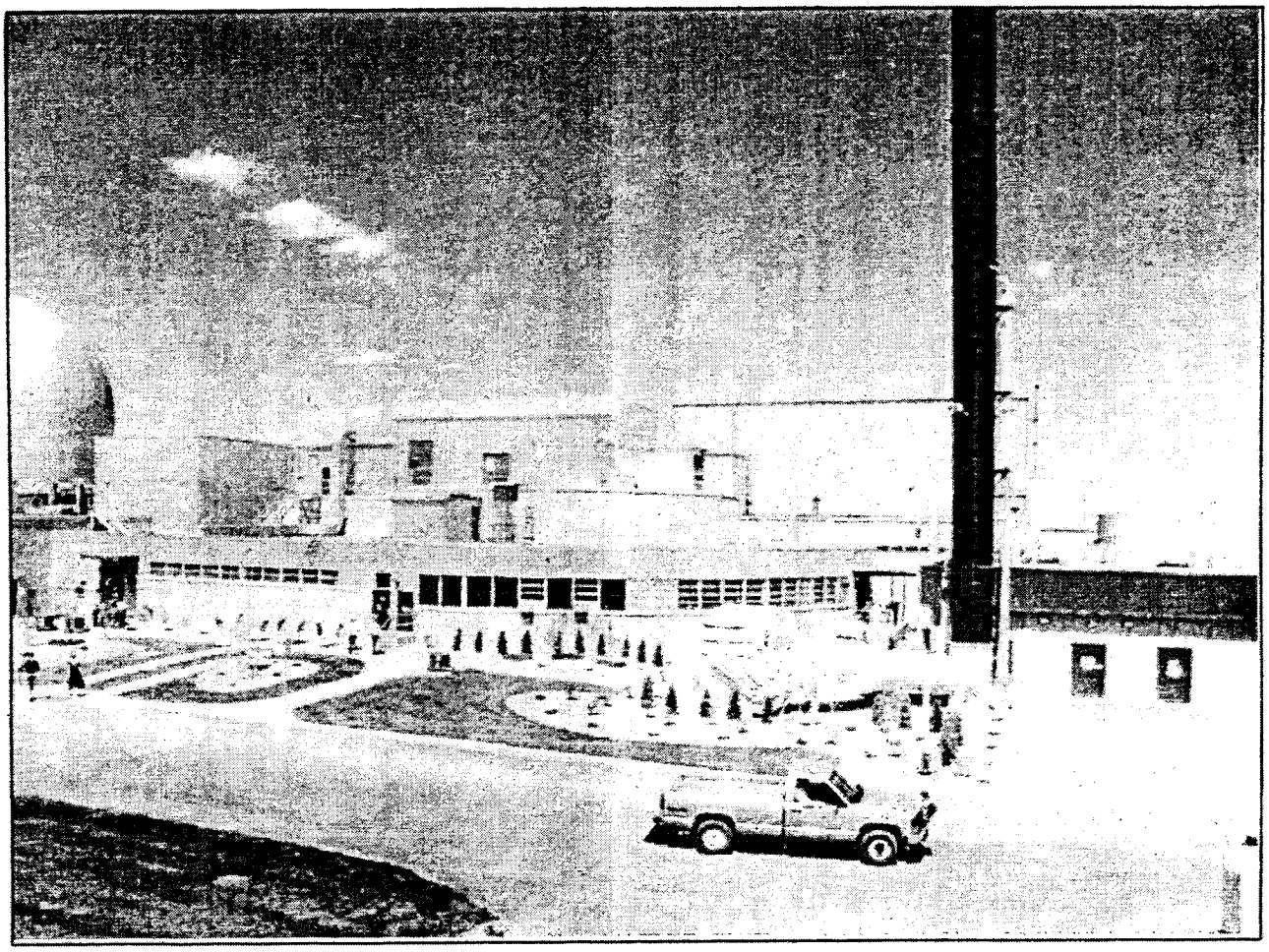

Figure 2–11 Fuel Conditioning Facility at ANL-W

2.4.1.2 Hot Fuel Examination Facility

The Hot Fuel Examination Facility is one of the proposed facilities for the treatment and management of sodium-bonded spent nuclear fuel. The Hot Fuel Examination Facility is a hot cell complex built in the early 1970s for the preparation and examination of irradiation experiments in support of a wide variety of programs and process demonstrations. A photographic view of the Hot Fuel Examination Facility is shown in **Figure 2–13**. A wide range of remote operations and examinations may be performed in this facility with its shielded cells, support areas, and equipment.

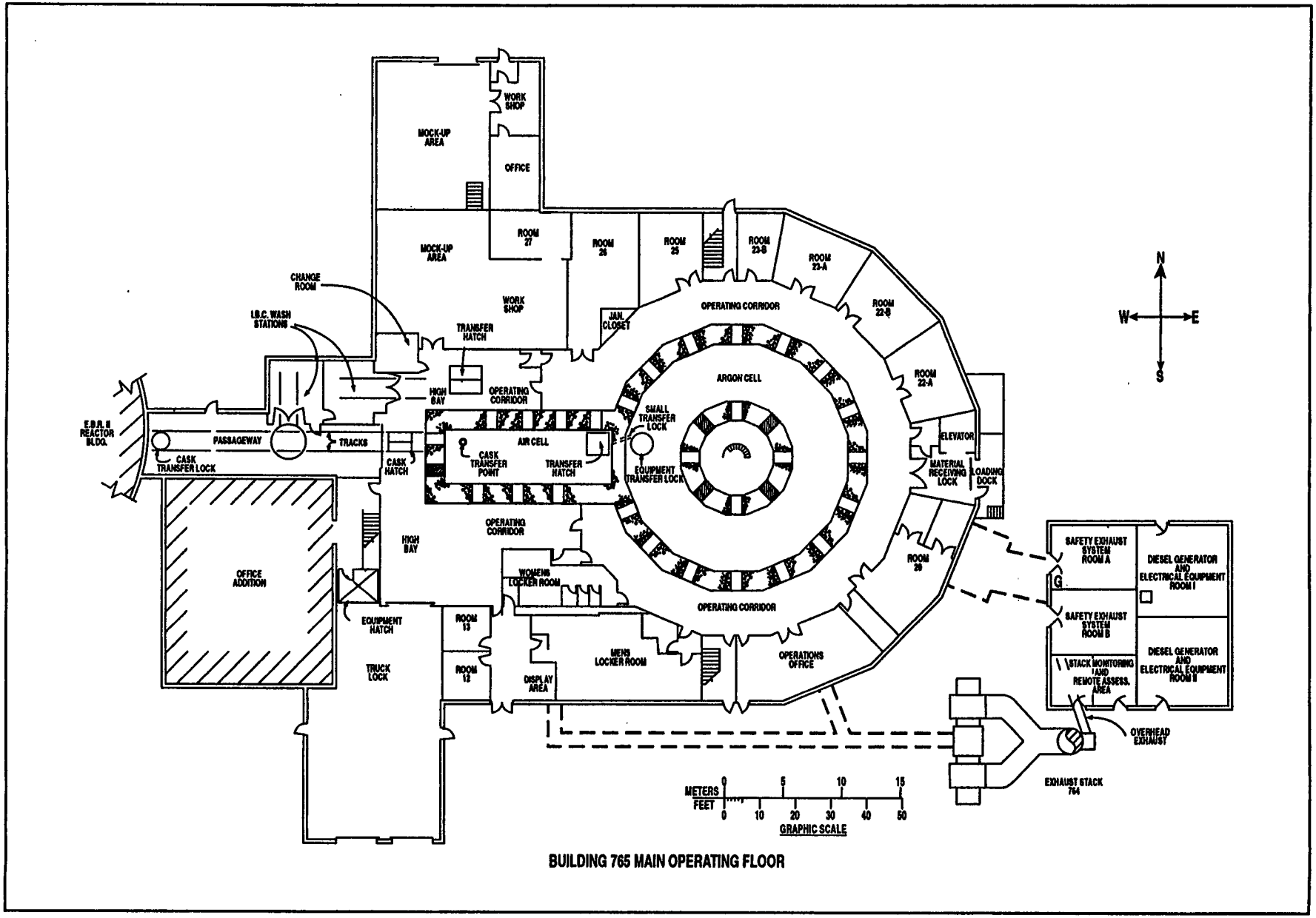

Figure 2–12 Main Floor Layout of the Fuel Conditioning Facility at ANL-W

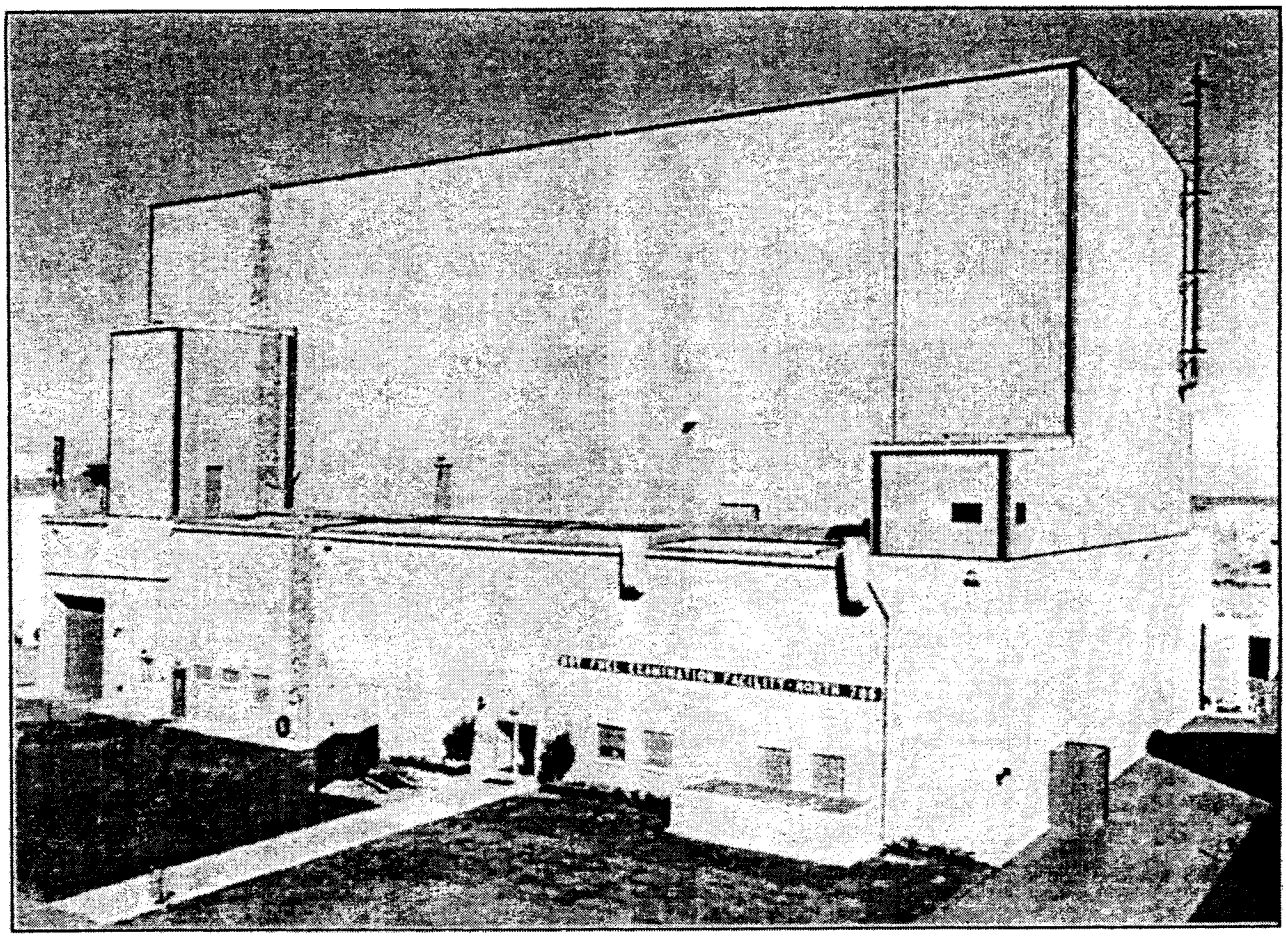

Figure 2–13 Hot Fuel Examination Facility at ANL-W

The Hot Fuel Examination Facility is designed to be adapted to a wide variety of programs and consists primarily of two adjacent shielded cells, the main cell and the decontamination cell, in a three-story building. The decontamination cell contains an air atmosphere. A schematic plan of the main floor of the Hot Fuel Examination Facility is shown in **Figure 2–14.** The main cell contains an argon atmosphere for work involving materials such as sodium, plutonium, and other materials which could react with air. Both cells are surrounded by 1.2-meter-thick (4-foot-thick), high-density concrete to protect workers from the high radiation levels present in the hot cells. There are 21 work stations in the Hot Fuel Examination Facility, all equipped with shielded windows and remote manipulators. All in-cell equipment is carefully designed to permit remote operation and maintenance. A truck lock is located at the west end of the cell complex. The truck lock is large enough to accommodate various trucks and fork lifts which are used to transport shielded casks used in the day-to-day operation of the facility. The facility has recently been modified to accept truck-sized spent nuclear fuel shipping casks.

A high bay area covering the entire cell complex and serviced by a 40-ton bridge crane provides access from the truck lock to the top of the cells for bottom-opening casks. This area contains repair rooms, change rooms, and an access room and provides space for clean equipment repair and mockup.

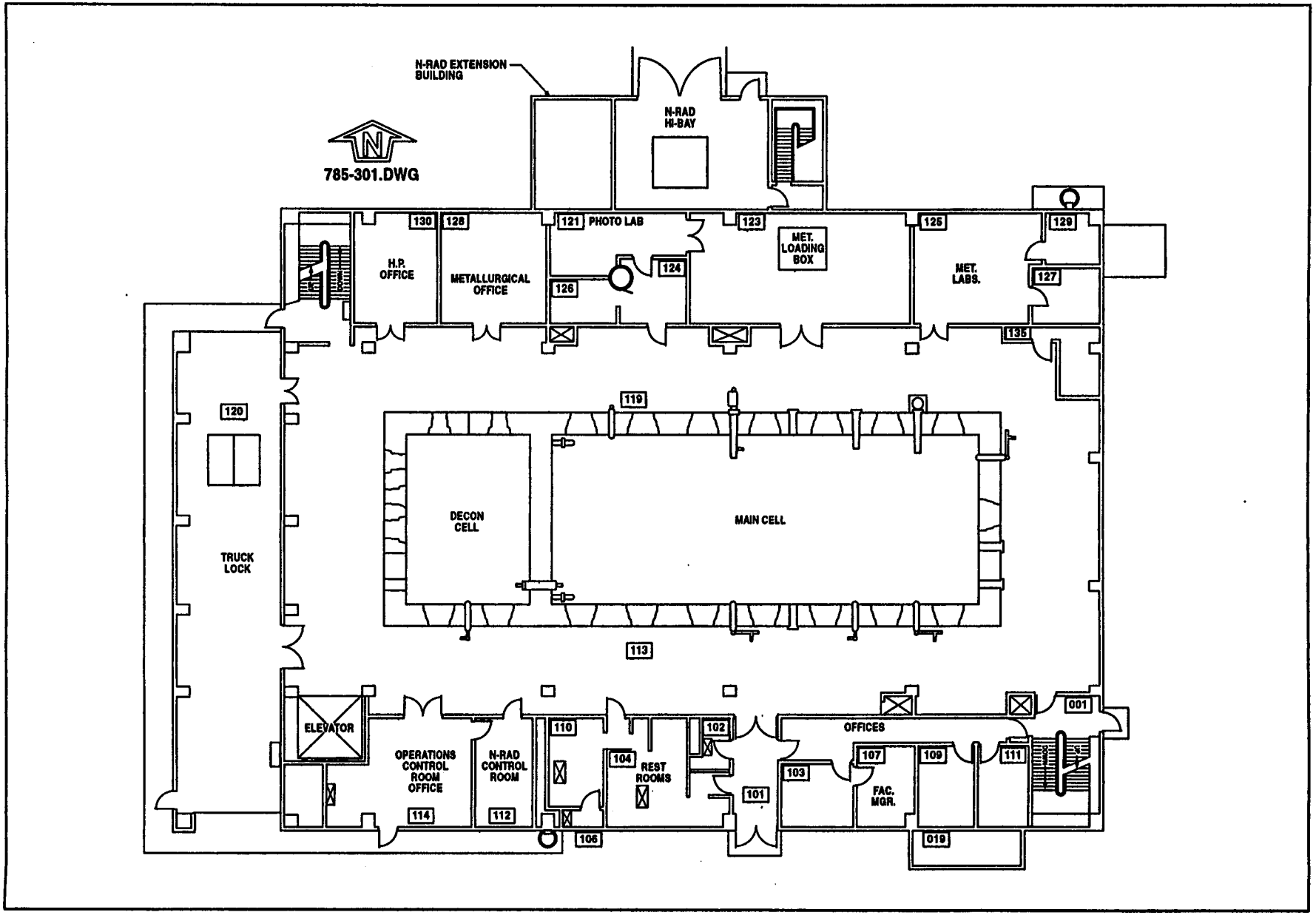

Figure 2–14 Main Floor Layout of the Hot Fuel Examination Facility at ANL-W

2.4.1.3 Zero Power Physics Reactor Materials Storage Building

The Zero Power Physics Reactor is currently shut down, but the facility is used for a number of projects, including a gas generation experiment. The Zero Power Physics Reactor Materials Storage Building, shown in **Figure 2–15**, is one of the primary storage facilities at ANL-W for uranium metal. Inventories of nuclear materials stored in this facility are maintained for conducting research as well as for storage, pending decisions for further disposition.

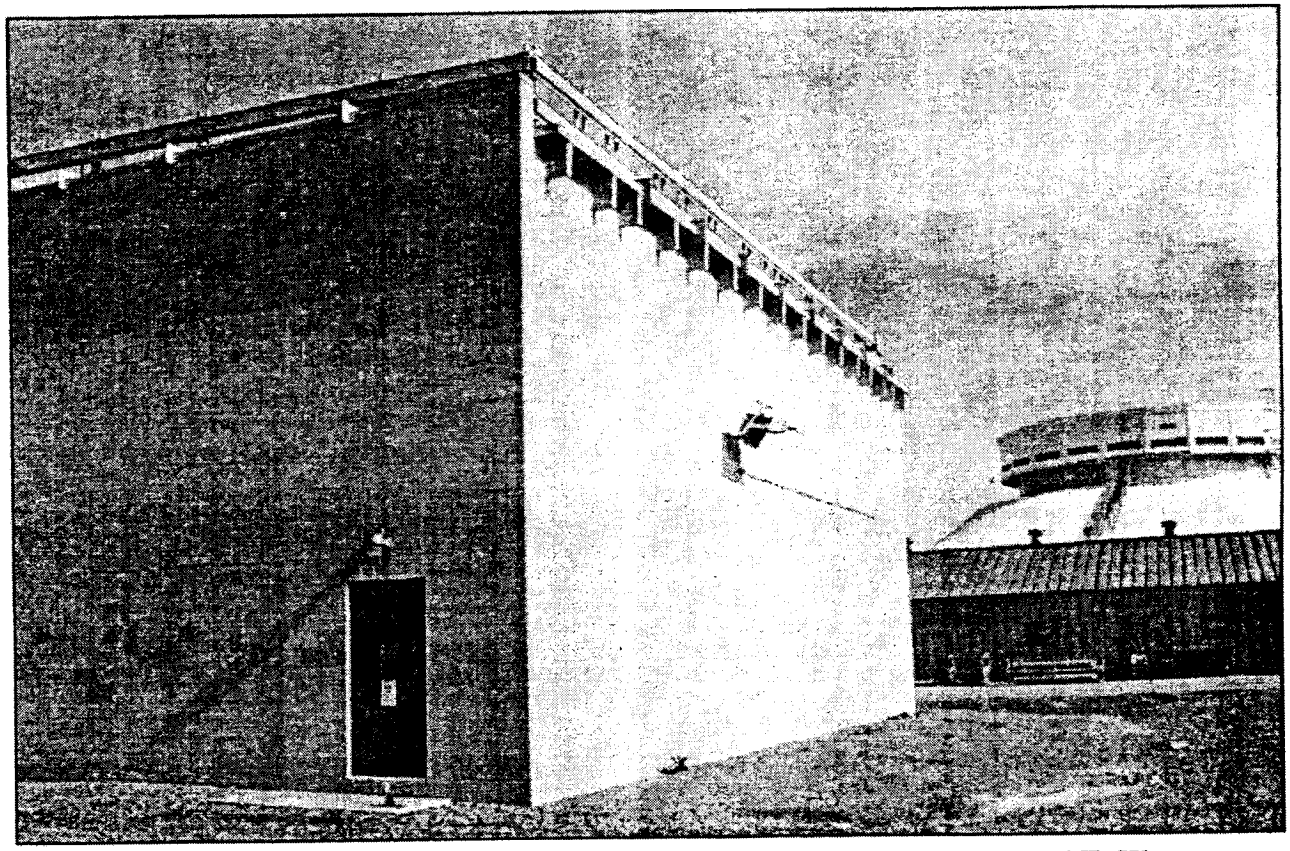

Figure 2–15 Zero Power Physics Reactor Materials Storage Building at ANL-W

2.4.1.4 Radioactive Scrap and Waste Facility

The Radioactive Scrap and Waste Facility, occupying about 1.6 hectares (4 acres), provides safe interim dry storage for spent nuclear fuel and waste generated from experiments. It is one of the facilities where sodium-bonded spent nuclear fuel currently is stored and the facility where high-level radioactive waste from the treatment of the fuel could be stored pending ultimate disposal. It is located underground and 0.8 kilometers (0.5 miles) northeast of the main ANL-W facilities within the ANL-W boundary. The Radioactive Scrap and Waste Facility looks somewhat like a large parking lot on the surface, as shown in **Figure 2–16**. The facility has a permit issued by the State of Idaho for interim storage of mixed waste regulated under RCRA. The Radioactive Scrap and Waste Facility provides protection against corrosion for the more than 1,000 underground steel liners available for waste storage of materials handled at ANL-W.

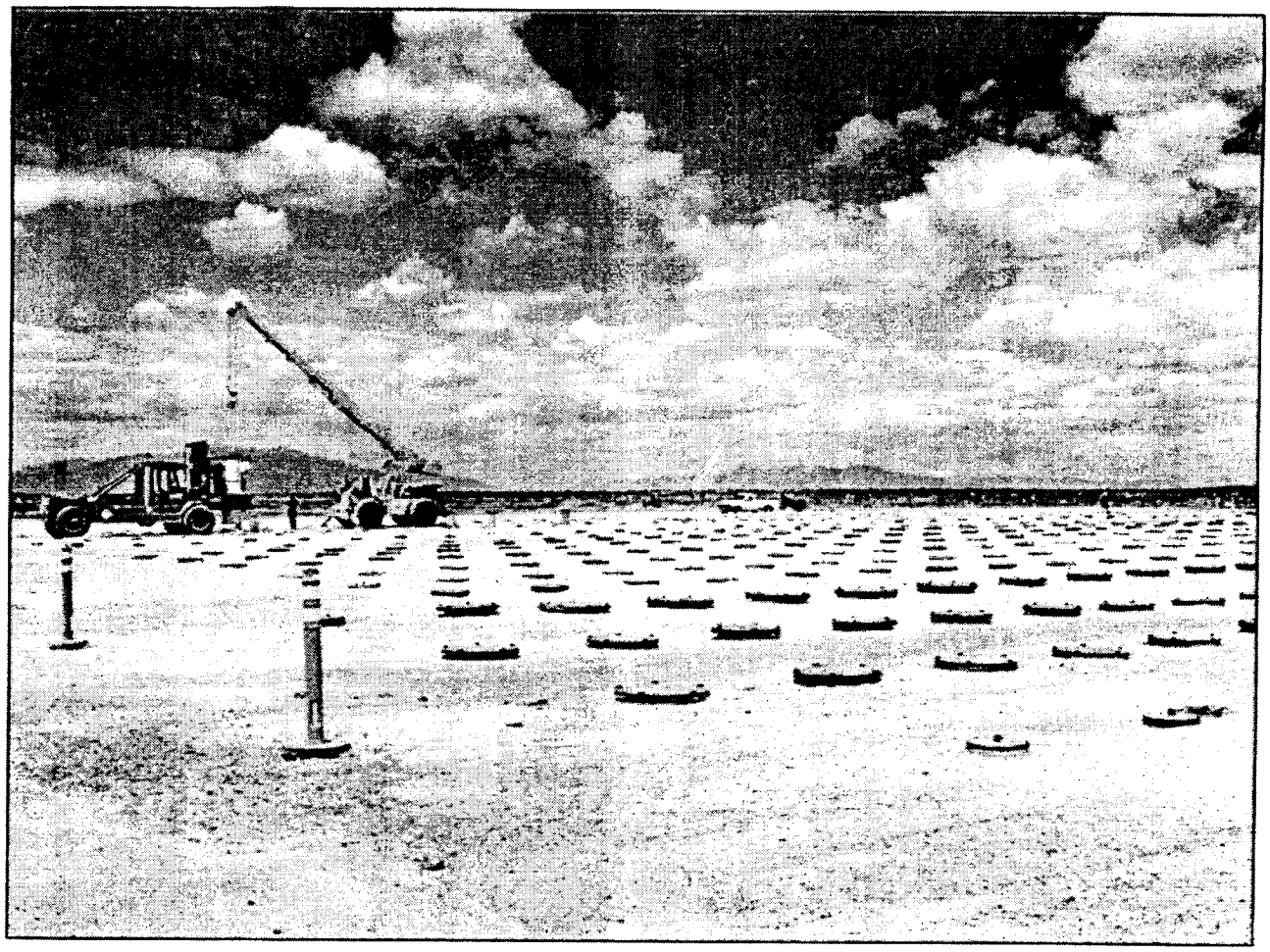

Figure 2–16 Radioactive Scrap and Waste Facility at ANL-W

2.4.2 INTEC at INEEL

INTEC is located 20 kilometers (12.4 miles) west-southwest of ANL-W. It is one of the sites where sodium-bonded spent nuclear fuel currently is stored. A photographic view of INTEC is shown in **Figure 2–17.** INTEC was constructed in the 1950s to reprocess spent nuclear fuel from government reactors. In 1992, DOE announced that it no longer would reprocess spent nuclear fuel. Current work at INTEC includes receiving and storing spent nuclear fuel, converting liquid sodium-bearing waste to granular solid, environmental restoration and decontamination and dismantling activities, and technology development. About 880 people currently work at INTEC. The facility would be used to continue storing sodium-bonded spent nuclear fuel and for packaging the treated sodium-bonded spent nuclear fuel in standardized canisters in preparation for transport and disposal in a geologic repository. However, because it has no hot cell with an inert gas atmosphere, it cannot be used for any sodium removal activities under the proposed action.

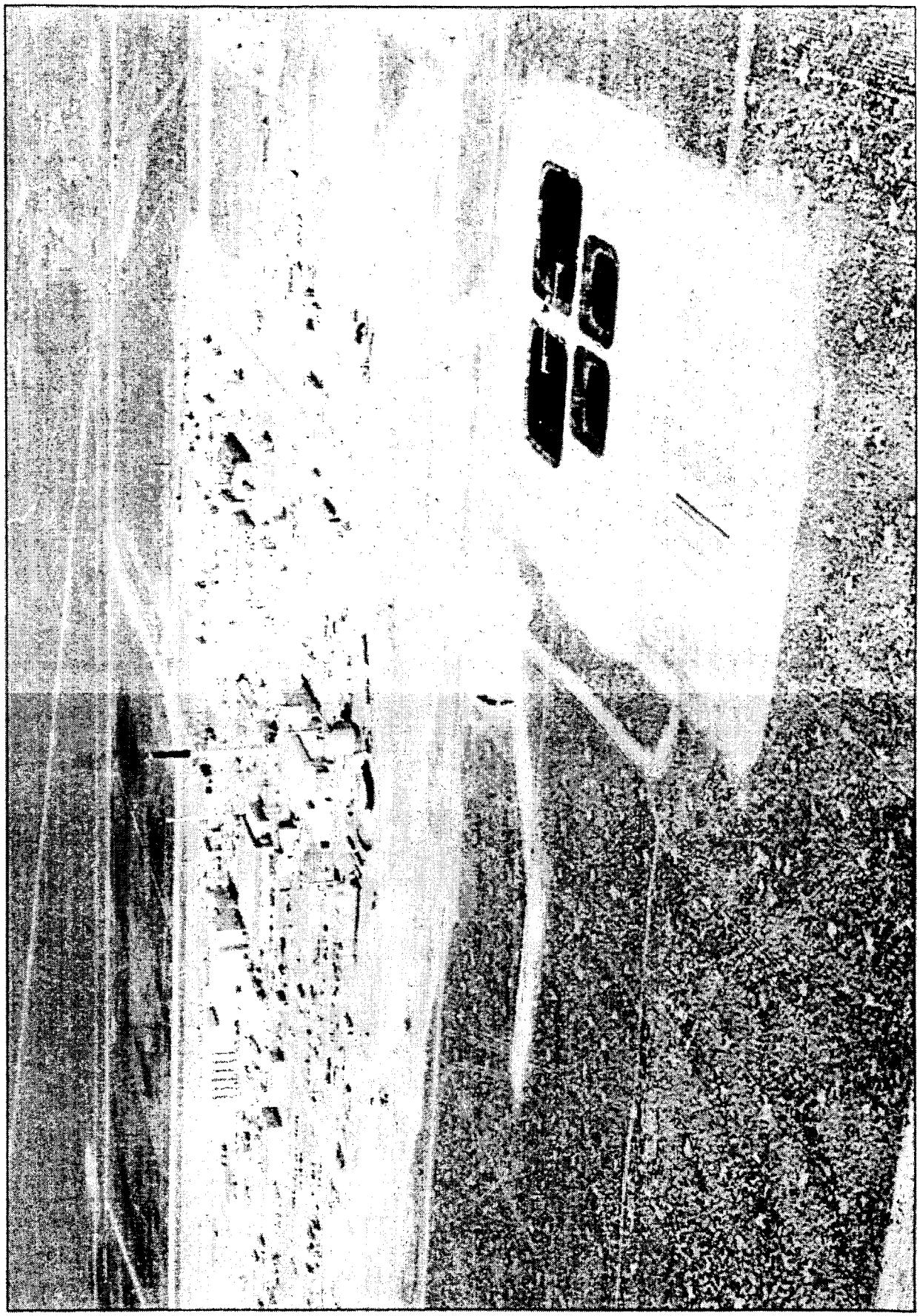

Figure 2–17 INTEC at INEEL

The primary facilities at INTEC include:

- The Fluorinel Dissolution Process and Fuel Storage Facility. This facility is divided into two parts, a spent nuclear fuel storage area and the Fluorinel Dissolution Facility. The storage area consists of six storage pools for storing nuclear fuel. Radioactive spent nuclear fuel is stored under about 11 million liters (3 million gallons) of water, which provide protective shielding and cooling. Eventually, all spent nuclear fuel will be removed from underwater storage pools and placed in a dry storage system and prepared for shipment to a repository.

  The spent nuclear fuel, from government-owned reactors, was formerly reprocessed at INTEC to recover reusable uranium. The Fluorinel Dissolution Facility includes an air atmosphere "hot cell" with 1.8-meter-thick (6-foot-thick) concrete walls where spent nuclear fuel was dissolved in an acid solution. With the end of reprocessing, uranium and hazardous materials were flushed from the Fluorinel Dissolution Facility. New missions for this facility are under consideration.

- The 603 Fuel Storage Building. This building houses both underwater pools and dry storage facilities for spent nuclear fuel. The pools were constructed in the 1950s and served as the primary spent nuclear fuel storage facility until the Fluorinel Dissolution Process and Fuel Storage Facility opened in 1984. Fuel in underwater storage at Building 603 is being transferred to the newer storage pools at the Fluorinel Dissolution Process and Fuel Storage Facility. Also located in the building is the Irradiated Fuel Storage Facility, which stores dry fuel that cannot be stored underwater. The Irradiated Fuel Storage Facility has 636 storage positions, with 297 in use. The majority of the spent nuclear fuel stored in this facility came from the Fort St. Vrain commercial reactor in Colorado.

- The New Waste Calcining Facility. This facility converted liquid high-level radioactive waste into a granular solid similar in consistency to dry laundry detergent. The liquid waste was drawn from underground storage tanks and sprayed into a vessel superheated by a mixture of kerosene and oxygen. Most of the liquid evaporated, while radioactive fission products adhered to the granular bed material in the vessel. The off-gases were treated and monitored before they were released to the environment, and the residual solids were transferred to large stainless steel structures encased in thick concrete vaults. This conversion process achieved an 8-to-1 volume reduction from liquid to solid. The same process currently is used to convert sodium-bearing waste to granular solid with a smaller volume reduction from liquid to solid.

- The Remote Analytical Laboratory. This laboratory is designed for the safe examination of radioactive samples to support the New Waste Calcining Facility mission and other INTEC operations. The facility includes a 356-square-meter (3,500-square-foot) air atmosphere hot cell with 90-centimeter (3-foot) concrete shield walls that allow remote examination of radioactive chemical samples.

- The INTEC 601/602 Processing Corridors. The processing corridors were used to extract highly enriched uranium from dissolved spent nuclear fuel during reprocessing and to solidify the recovered highly enriched uranium for shipment off site. Built in the 1950s, these facilities were to be replaced by the Fuel Processing Restoration Project. Because DOE decided to discontinue reprocessing, these facilities have been flushed to remove uranium and hazardous materials. The Fuel Processing Restoration Project, about 40 percent complete when construction stopped in 1992, was discontinued in a manner that preserves the facility for possible use in future research and development missions at INTEC.

- The High-Level Radioactive Waste Tank Farm. The waste tank farm includes 11 underground stainless steel storage tanks used to store the radioactive liquid waste generated during the reprocessing of spent nuclear fuel and plant decontamination work. DOE is evaluating options for treating the remaining Tank Farm liquid waste in the *Idaho High-Level Waste and Facilities Disposition Draft Environmental Impact*

Statement, which was issued for public comment in December 1999 (DOE 1999b). The underground tanks are encased in concrete vaults which have sumps and leak detectors. One tank is always kept empty for use as a transfer backup should a problem develop with one of the other 10 tanks. The tanks are corrosion-resistant, and no leakage has been detected. Some leaks from transfer lines outside the tanks have occurred, however, and the contaminated soil is scheduled for environmental cleanup.

2.4.3 SRS

SRS was constructed during the early 1950s to produce the basic materials used in the fabrication of nuclear weapons, primarily tritium and plutonium-239. Five reactors were built on the site. The reactors produced nuclear materials by irradiating target materials with neutrons. Also built were support facilities including two chemical separation plants, a heavy water extraction plant, a nuclear fuel and target fabrication facility, and waste management facilities.

Irradiated materials were moved from the reactors to the two chemical separation facilities—the next step in the production process. In these facilities, known as "canyons," the irradiated fuel and target assemblies were processed chemically to separate useful products from waste. After refinement, some nuclear materials were shipped to other DOE sites for final use.

SRS has adjusted through the years to meet changing defense requirements. All five of the original SRS production reactors are permanently shut down. While production of new tritium will not be necessary for several years, recycling and reloading of tritium to maintain nuclear weapons' reliability is a continuing site mission.

DOE currently uses the F-Canyon chemical separation facility and the FB-Line to stabilize spent nuclear fuel, as described in the *Final Environmental Impact Statement, Interim Management of Nuclear Materials* (DOE 1995b). The *Savannah River Site Spent Nuclear Fuel Management Final Environmental Impact Statement* (DOE 2000) will be used to help determine the most appropriate final disposition option for spent nuclear fuel currently assigned to SRS.

Weapons material production at SRS has produced unusable by-products such as high-level radioactive waste. The high-level radioactive waste, approximately 35 million gallons, is stored in waste tanks on site. The Defense Waste Processing Facility will bond the radioactive elements in borosilicate glass, a stable form for disposal.

2.4.3.1 F-Canyon

The F-Canyon at SRS could chemically separate uranium from fission products in blanket spent nuclear fuel using the PUREX process. A photographic view of the F-Canyon complex is shown in **Figure 2–18.** The canyon facilities use radiochemical processes for the separation and recovery of plutonium and uranium isotopes. Historically, F-Canyon recovered plutonium-239 and uranium-238 from irradiated natural or depleted uranium.

The Canyon building is a reinforced concrete structure, 254 meters (835 feet) long by 37 meters (122 feet) wide by 20 meters (66 feet) high. It houses large equipment (tanks, process vessels, evaporators, etc.) used in the chemical separation processes.

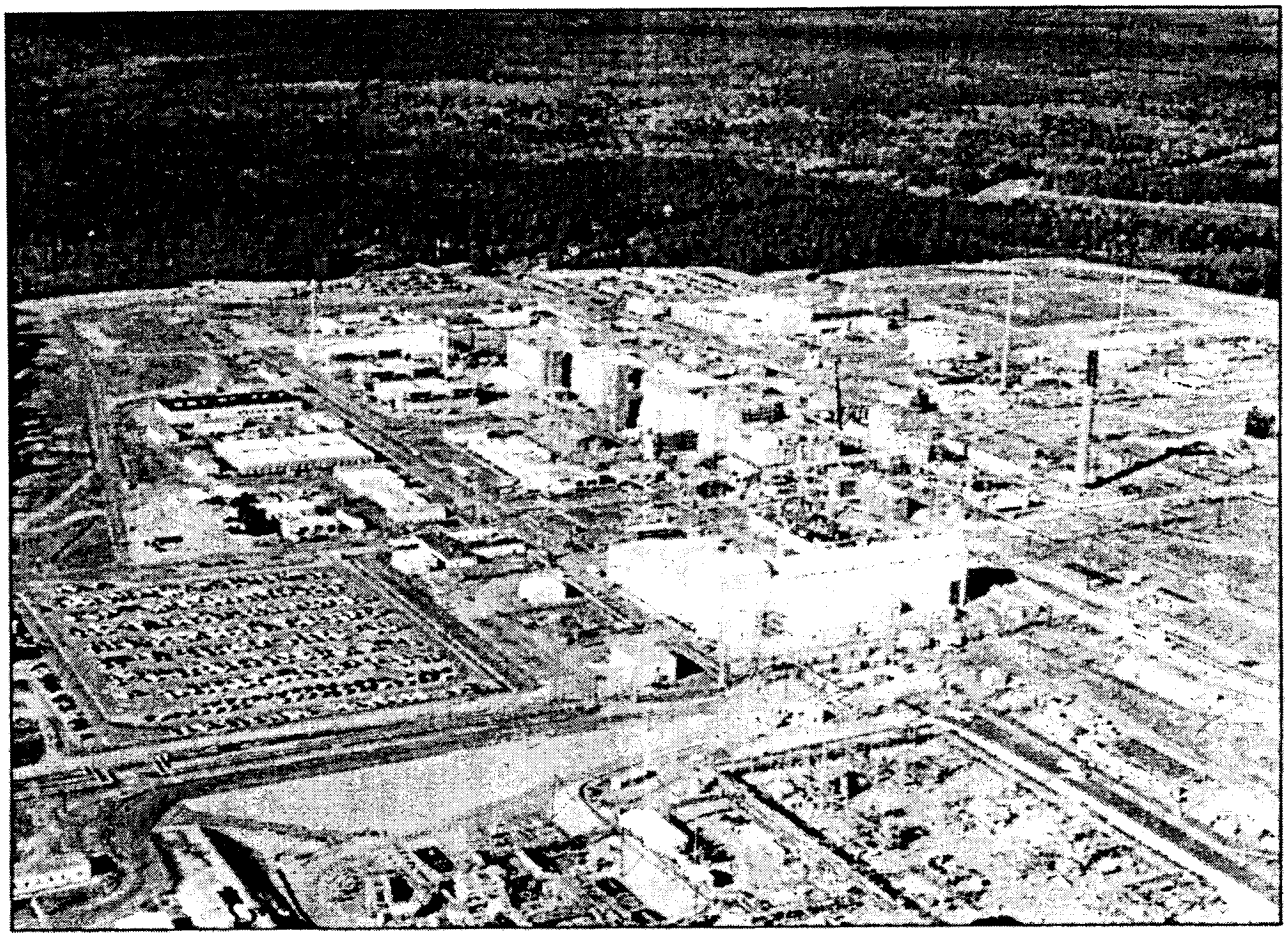

Figure 2–18 The F-Canyon Complex at SRS

The F-Canyon facility actually contains two canyons, the hot canyon and the warm canyon, as shown in **Figure 2–19**. The two canyons are parallel and separated by a center section, which has four floors. The center section contains office space, the control room for 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 and in the center section from radiation. The final steps of the chemical separation process, which generally involve lower radiation levels, occur in the warm canyon.

2.4.3.1.1 FB-Line

The FB-Line, located on top of the F-Canyon, historically converted plutonium nitrate solution produced in the F-Canyon to plutonium-239 metal buttons. Solutions from the F-Canyon are concentrated and purified in the FB-Line. The plutonium then is precipitated, filtered, dried, and finally reduced to a metallic form called a button. The button is about the size of a hockey puck. Processing equipment is enclosed in glove boxes so that employees and operating areas are not exposed to the radioactive material. Some of these operations are automated. The FB-Line also recycles plutonium scrap generated from facility operations and offsite sources. In September 1997, the FB-Line began a new plutonium packaging process. This process places stabilized plutonium in rugged, welded stainless steel cans. DOE also has determined that the FB-Line should be used to stabilize plutonium.

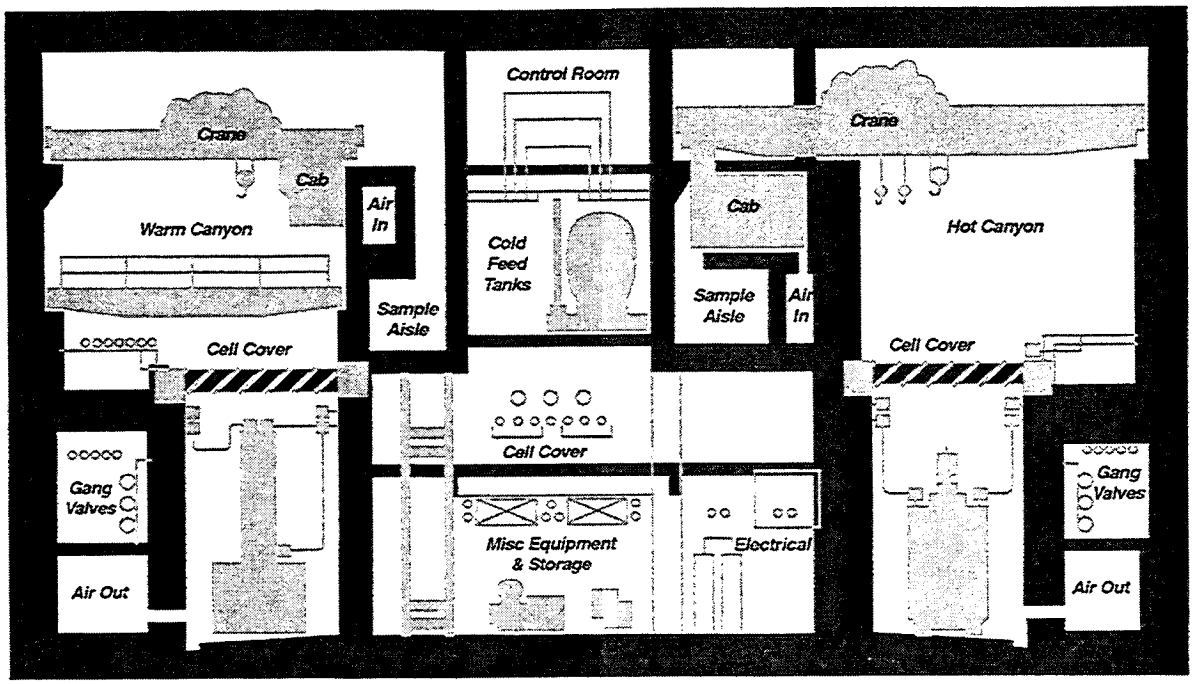

Figure 2–19 F-Canyon Building Sections (Hot Canyon and Warm Canyon) at SRS

2.4.3.2 Building 105-L

Building 105-L is the SRS facility where installation of a melt and dilute process for treating spent nuclear fuel is proposed. Building 105-L is part of the currently shut-down L-Reactor complex at SRS. The L-Reactor was built in the early 1950s to produce nuclear materials for national defense. In 1988, DOE shut the reactor down for safety upgrades. At the completion of the upgrades, the reactor was not restarted. A photographic view of Building 105-L is shown in **Figure 2–20**. In 1993, DOE ended the reactor's materials production mission. The current mission of this facility is to store reactor components and other radioactive materials in the disassembly basin; receive and store foreign and domestic research reactor fuel in the disassembly basin; decontaminate shipping casks in the Building 105-L stack area; store contaminated moderators in tanks or drums; and compact low-level radioactive waste in a compactor. DOE maintains the structures, systems, and components necessary to perform these missions, but has de-energized, drained, or otherwise deactivated many others.

Building 105-L has space potentially suitable for installation of a melt and dilute process (DOE 2000). The space includes the process room and crane maintenance area. The process room, a shielded area situated above the reactor tank, formerly provided access to the reactor by means of a charge and discharge machine for handling reactor fuel assemblies. An overhead crane services the area. The crane maintenance area, connected to the process room by a shielded crane wash area, allows hands-on maintenance of the fuel assembly transfer systems. The Building 105-L stack area would be used to unload shipping casks from their containers and to decontaminate empty shipping casks.

In the *Savannah River Site Spent Nuclear Fuel Management Final Environmental Impact Statement*, (DOE 2000), DOE identified melt and dilute as one of the preferred methods of treating spent nuclear fuel at SRS. To implement the melt and dilute technology, DOE would construct a melt and dilute facility in the existing Building 105-L at SRS and build a dry storage facility in L-Area, near Building 105-L.

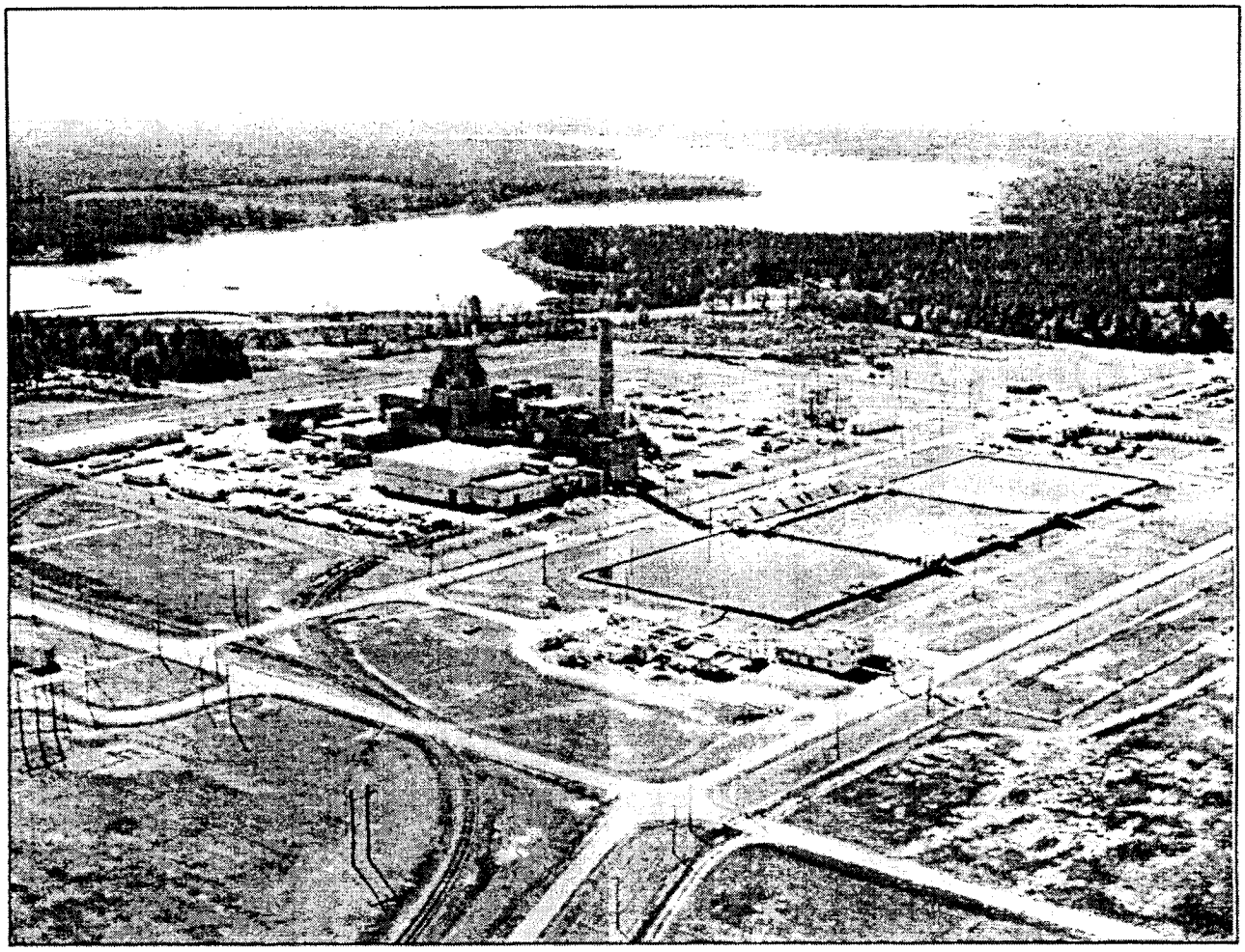

Figure 2–20 L-Reactor Complex at SRS

DOE expects the melt and dilute option would be relatively simple to implement in Building 105-L. The major technical issue in implementing this technology would be the design of an off-gas system to capture volatilized fission products. Preliminary engineering studies indicate that the system could be designed using proven approaches for managing off-gases. The impacts from the construction of a melt and dilute facility at SRS's Building 105-L are addressed in the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000).

2.4.3.3 Defense Waste Processing Facility

The Defense Waste Processing Facility, located in the S-Area, converts high-level radioactive liquid waste currently stored at SRS into a solid glass form suitable for long-term storage and disposal. A photographic view of the Defense Waste Processing Facility is shown in **Figure 2–21**. This process, called "vitrification," immobilizes high-level radioactive liquids into a more stable form suitable for disposal in a geologic repository. About 125 million liters (34 million gallons) of high-level liquid radioactive waste currently is stored in 49 underground carbon steel tanks at SRS. This waste has about 480 million curies of radioactivity, and requires permanent isolation from the environment.

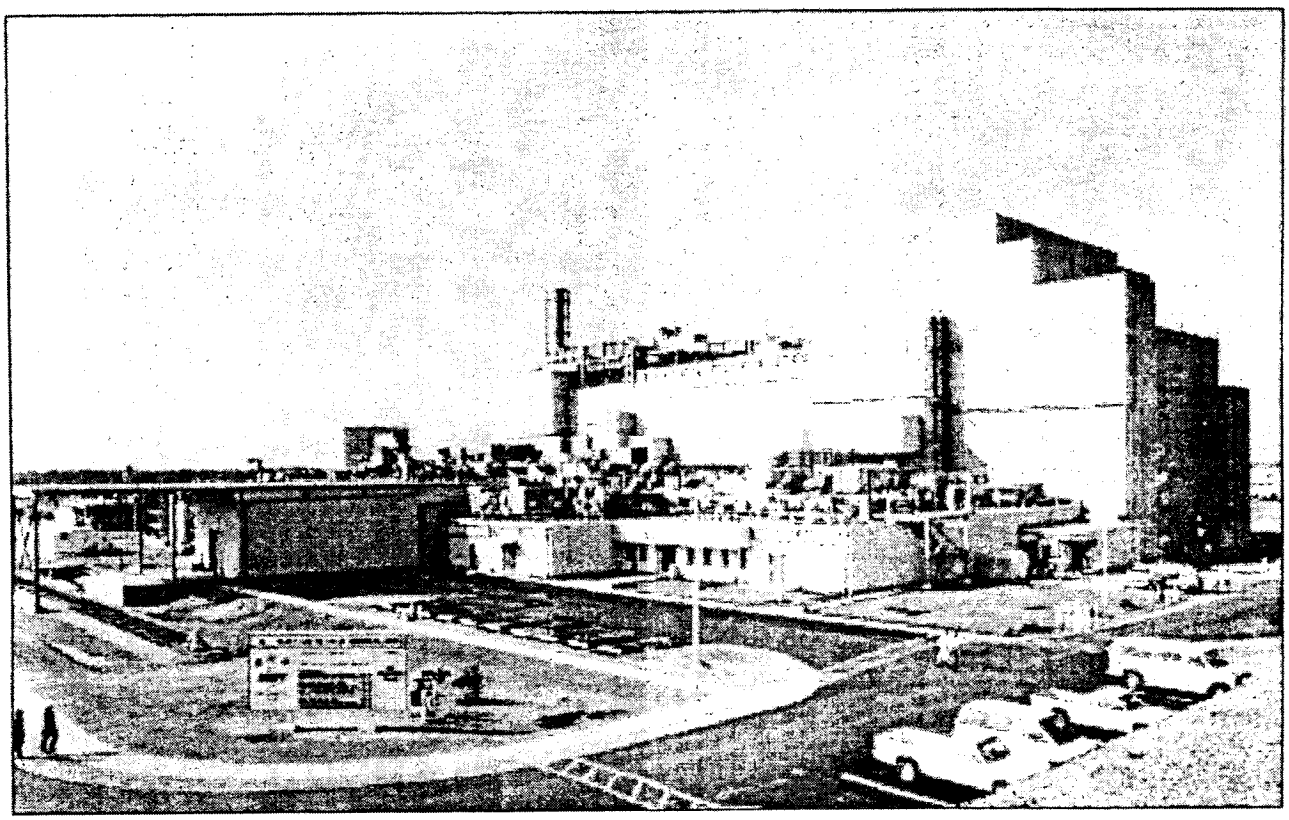

Figure 2–21 Defense Waste Processing Facility at SRS

Construction of the Defense Waste Processing Facility began in 1983. Changing environmental requirements; major safety upgrades and process modifications; and a "waste qualification" test to demonstrate that the glass form meets all environmental and operational requirements for long-term storage were required before system testing began in 1990. The Defense Waste Processing Facility successfully completed its waste qualification testing in late 1995 and began operating in March 1996.

The Defense Waste Processing Facility treats the highly radioactive material removed from the original waste. In this process, a sand-like borosilicate glass is mixed with the waste and sent to the plant's steel and ceramic melter. In the melter, electricity is used to heat the waste/borosilicate glass mixture until molten. This molten glass-waste mixture is poured in a pencil-thin stream into stainless steel canisters to cool and harden. Each canister is approximately 3 meters (10 feet) tall and 0.6 meters (2 feet) in diameter; it takes approximately 24 hours to fill one canister. A filled Defense Waste Processing Facility canister weighs about 2,270 kilograms (5,000 pounds). The exterior of each canister is blasted with borosilicate glass to remove contamination, then welded shut after a plug has been rammed into place.

A specially designed "Shielded Canister Transporter" moves each sealed canister, one at a time, from the Defense Waste Processing Facility to the temporary storage building adjacent to the facility. This transporter is a two-wheel drive vehicle powered by redundant diesel engines. It has a center module with a shielding cask, floor plug cavity, and associated cask lifting equipment.

At the storage building, canisters are lowered by the transporter into an underground reinforced concrete vault containing 2,286 individual canister supports. This seismically qualified storage vault can hold, at current Defense Waste Processing Facility production flow, canisters from about 8 to 10 years of processing. More storage buildings can be built according to the need for storage space. The canisters are stored at SRS until a Federal repository can be established.

2.5 ALTERNATIVES EVALUATED

As discussed in Section 2.1, the proposed action is to treat and manage sodium-bonded spent nuclear fuel. The alternatives under the proposed action are illustrated in **Figure 2–22** and are addressed below. Although each alternative addresses both driver and blanket spent nuclear fuel, DOE will consider the driver and blanket spent nuclear fuel, as well as the EBR-II and Fermi-1 blanket spent nuclear fuel, separately in identifying a preferred alternative and any subsequent Record of Decision. In other words, DOE is considering all combinations of technologies, options, and fuel types, not only the specific combinations that are explicitly discussed in the EIS. For example, "no action" may be chosen for the driver spent nuclear fuel and "melt and dilute at SRS" for the blanket spent nuclear fuel.

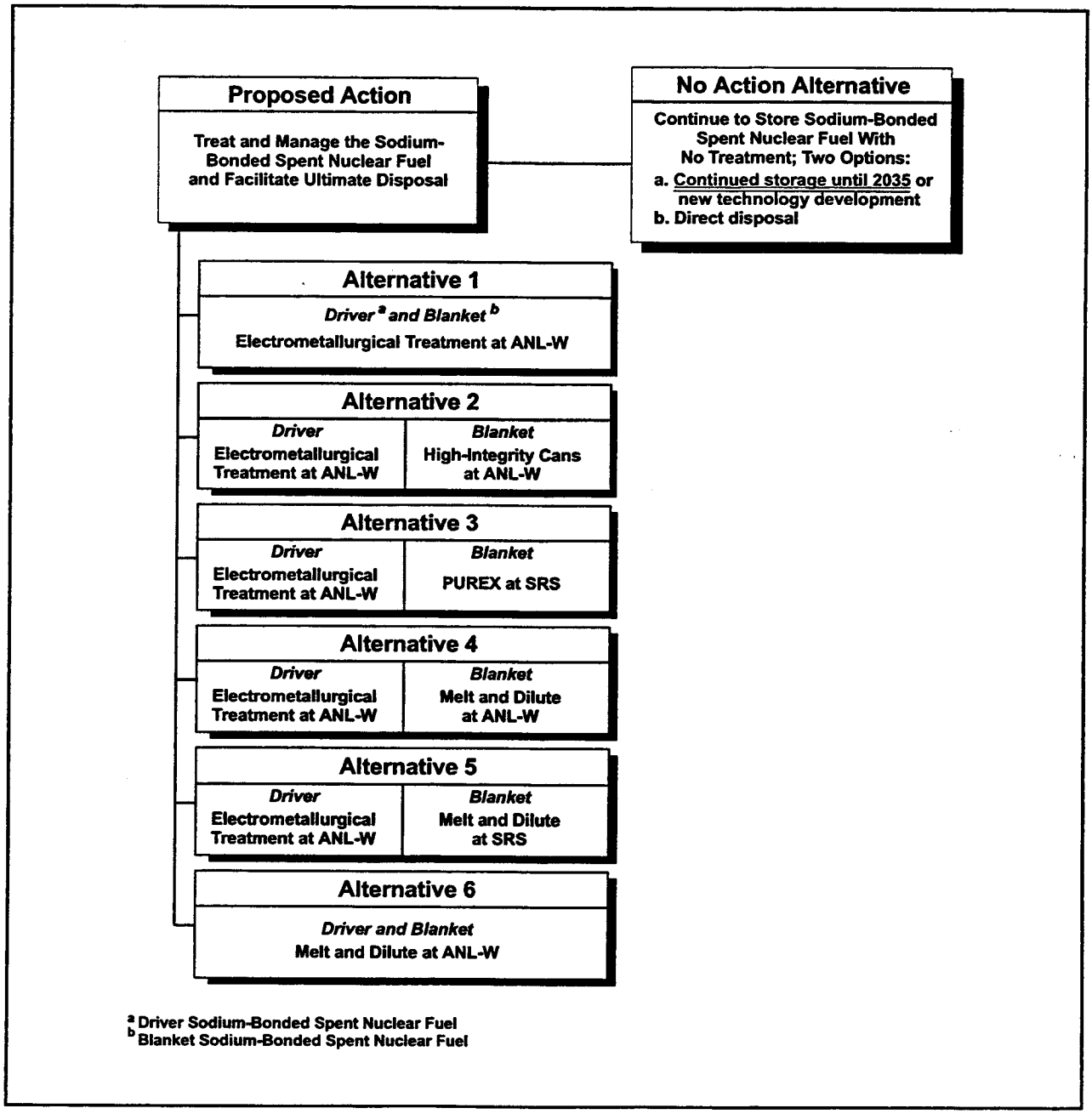

Figure 2–22 Proposed Action and Alternatives

2.5.1 No Action Alternative

Under the No Action Alternative, all or part of the sodium-bonded spent nuclear fuel would not be treated (no sodium would be removed), except for stabilization activities that may be necessary to prevent potential degradation of some of the spent nuclear fuel. Under the No Action Alternative, two options were analyzed: (1) the sodium-bonded spent nuclear fuel would continue to be stored until 2035 at its current location subject only to activities dictated by the amended Record of Decision (61 FR 9441) for the Programmatic Spent Nuclear Fuel EIS (DOE 1995a) and other existing site-specific NEPA documentation or until a technology currently dismissed as an unreasonable alternative because it is less mature (e.g., GMODS or plasma arc) is developed; and (2) the sodium-bonded spent nuclear fuel would be disposed of directly in a geologic repository without treatment. The fuel would be packaged in high-integrity cans without sodium removal. Under the latter option, the sodium-bonded spent nuclear fuel at INTEC would be transported to ANL-W for packaging. Both options would require the installation of some new waste handling equipment. As discussed in Section 2.3.8, the direct disposal option would not meet current DOE or NRC repository acceptance criteria requirements.

A fundamental assumption made under the No Action Alternative is that the sodium-bonded spent nuclear fuel eventually will be disposed of in a manner similar to the rest of the spent nuclear fuel owned by DOE and within the time period considered over which institutional controls could reliably be assumed to be in effect. In the event that the sodium-bonded spent nuclear fuel has not been treated before 2035, the temporarily stored fuel will be removed from the State of Idaho by the year 2035. The environmental impact of the removal of untreated sodium-bonded spent nuclear fuel would be evaluated in a separate NEPA document. The continued storage of untreated sodium-bonded spent nuclear fuel in the State of Idaho or elsewhere, beyond time periods for which institutional controls could reliably be assumed to be in effect, could lead to significant impacts to the environment and the health and safety of the public from radioactive releases caused by the gradual degradation of the fuel and its containment.

In selecting the No Action Alternative, DOE could actively pursue research and development of another treatment technology including, for example, the GMODS and plasma arc methods. These methods offer the potential for treating both driver and blanket spent nuclear fuel. They do not involve separation of uranium or plutonium, and the treatment product is expected to be suitable for disposal in a geologic repository. Reasons for not including these methods among the reasonable alternatives under the proposed action are provided in Section 2.6.

2.5.2 Alternative 1: Electrometallurgically Treat Blanket and Driver Fuel at ANL-W

Under this alternative, all sodium-bonded spent nuclear fuel (both driver and blanket, approximately 60 metric tons of heavy metal) would be treated at ANL-W using the electrometallurgical treatment process. **Figure 2–23** illustrates the steps of the process under Alternative 1.

The sodium-bonded spent nuclear fuel (driver and blanket) from ANL-W (the Radioactive Scrap and Waste Facility and the Hot Fuel Examination Facility) would be transported directly to the Fuel Conditioning Facility for electrometallurgical treatment. Spent nuclear fuel currently stored at INTEC would be transported to the Hot Fuel Examination Facility. This is necessary because only the Hot Fuel Examination Facility at ANL-W is capable of accepting spent nuclear fuel transportation casks. At the Hot Fuel Examination Facility, the fuel would be separated from the assembly hardware and packaged and transferred to the Fuel Conditioning Facility for electrometallurgical treatment. The separated hardware would be packaged and managed as low-level radioactive waste.

After treatment, the low-enriched uranium by-product from the cathode processing would be metal-casted at the Fuel Conditioning Facility and transferred to the Zero Power Physics Reactor Materials Storage Building

for storage. The cladding hulls remaining at the anode would be packaged and transferred to the Hot Fuel Examination Facility for metal casting into high-level radioactive waste and transferred to the Radioactive Scrap and Waste Facility for storage. The electrorefiner salt containing the fission products, sodium, and transuranic elements would be transferred in metal cans back to the Hot Fuel Examination Facility where the ceramic waste would be produced. The ceramic waste cylinders would be packaged and transferred to the Radioactive Scrap and Waste Facility for storage. Implementing this alternative at the Fuel Conditioning Facility and the Hot Fuel Examination Facility would require the installation of some new waste handling equipment at the facilities. Electrometallurgical treatment of the sodium-bonded spent nuclear fuel at ANL-W could start as early as the year 2000, and would require approximately 13 years to process all fuel. Driver spent nuclear fuel alone would require approximately 7 years.

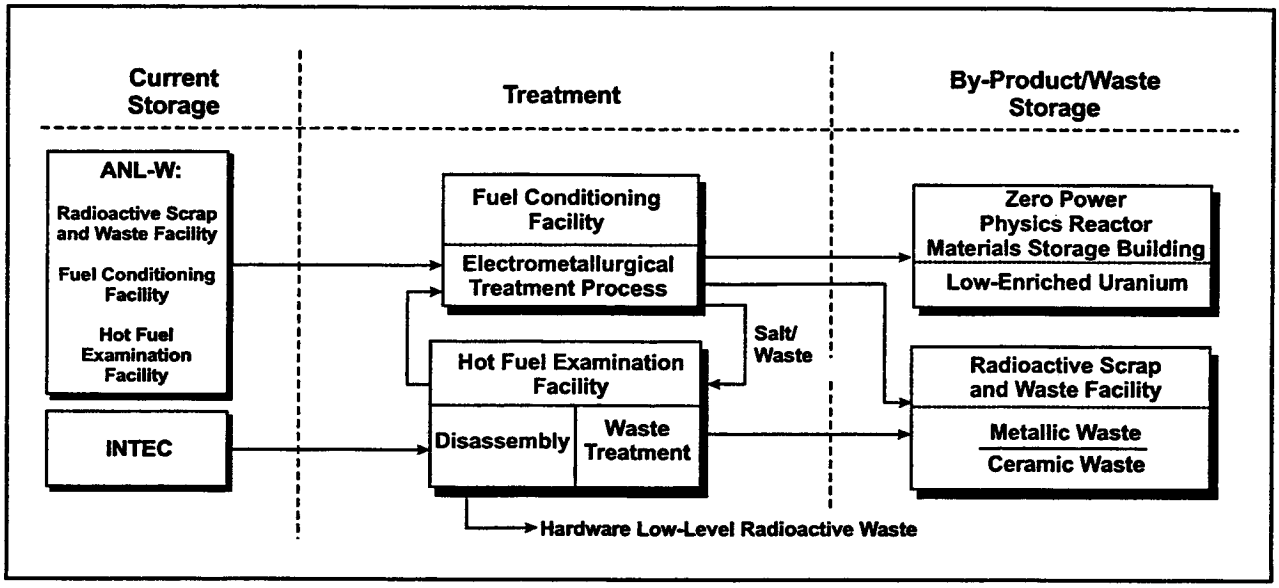

Figure 2–23 Schematic for Driver and Blanket Spent Nuclear Fuel Treatment in Alternative 1

2.5.3 Alternative 2: Clean and Package Blanket Fuel in High-Integrity Cans and Electrometallurgically Treat Driver Fuel at ANL-W

Under this alternative, the blanket spent nuclear fuel elements (approximately 57 metric tons of heavy metal) would be packaged in high-integrity stainless steel cans at ANL-W after the sodium has been removed without decladding, as discussed in Section 2.3.9. Removal of the sodium from the sodium-bonded blanket spent nuclear fuel would take place at the Hot Fuel Examination Facility at ANL-W. The packaging in high-integrity cans would take place in the same facility. The high-integrity cans would be transferred to the Radioactive Scrap and Waste Facility for storage. **Figure 2–24** illustrates the steps of the process for the blanket sodium-bonded spent nuclear fuel.

The sodium-bonded driver spent nuclear fuel (approximately 3 metric tons of heavy metal) would be treated using the electrometallurgical treatment process as described in Section 2.5.2 for Alternative 1.

Implementing this alternative at the Hot Fuel Examination Facility would require the installation of equipment for sodium removal activities. In addition, some new waste handling equipment would be needed for the electrometallurgical treatment of the driver sodium-bonded spent nuclear fuel.

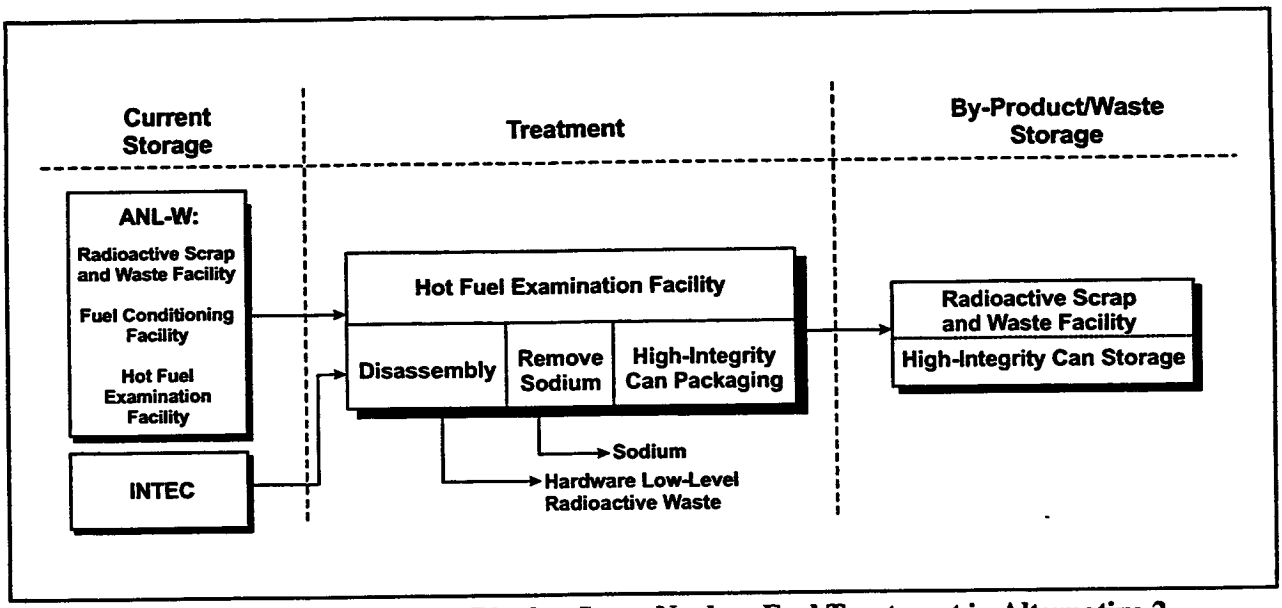

Figure 2–24 Schematic for Blanket Spent Nuclear Fuel Treatment in Alternative 2

Packaging the blanket spent nuclear fuel in high-integrity cans could start by approximately 2003. It would take approximately six years to complete. Electrometallurgical treatment of the driver spent nuclear fuel would start in 2000 and would be completed in approximately seven years.

2.5.4 Alternative 3: Declad and Clean Blanket Fuel and Electrometallurgically Treat Driver Fuel at ANL-W; PUREX Process Blanket Fuel at SRS

Under this alternative, the blanket spent nuclear fuel pins (approximately 57 metric tons of heavy metal) would be packaged in aluminum cans and shipped to SRS for treatment using the PUREX process at the SRS F-Canyon facility. The blanket spent nuclear fuel pins would be separated from the cladding and cleaned to remove the metallic sodium at ANL-W, as discussed in Section 2.3.9.

The decladding of the sodium-bonded blanket spent nuclear fuel and sodium removal would take place at the Hot Fuel Examination Facility at ANL-W. Equipment for decladding and sodium removal would need to be installed for this purpose. After decladding and sodium removal, the blanket spent nuclear fuel pins would be packaged and stored temporarily at the Hot Fuel Examination Facility to await shipment to SRS.

At SRS, the cans containing blanket spent nuclear fuel pins would be unpacked at the F-Canyon facility before treatment using the PUREX process. No modifications to that facility would be needed. Waste from the process containing the fission products and transuranic isotopes other than plutonium would be transferred to the Defense Waste Processing Facility where it would be converted to borosilicate glass logs and stored pending ultimate disposal. Approximately 260 kilograms (572 pounds) of separated plutonium would be immobilized using the can-in-canister technology at SRS for eventual geologic repository disposal in accordance with the Record of Decision (75 FR 1608) for the *Surplus Plutonium Disposition Final Environmental Impact Statement* (DOE 1999c). Depleted uranium would be transferred to a storage yard for depleted uranium at the site. **Figure 2–25** illustrates the process steps for the blanket spent nuclear fuel at ANL-W and SRS.

Considering the commitment of F-Canyon to other DOE missions, PUREX processing of the blanket spent nuclear fuel would start no earlier than 2005, and would last less than one year. Decladding and sodium

removal activities at ANL-W would not start earlier than 2003. Therefore, these activities would determine the length of the process.

The sodium-bonded driver spent nuclear fuel (approximately 3 metric tons of heavy metal) would be treated at ANL-W using the electrometallurgical treatment processes described in Section 2.5.2 for Alternative 1. As in the case of Alternative 2, electrometallurgical treatment of the driver spent nuclear fuel could start in 2000 and could be completed in approximately seven years. The process steps for the treatment of the driver spent nuclear fuel are shown in Figure 2–23.

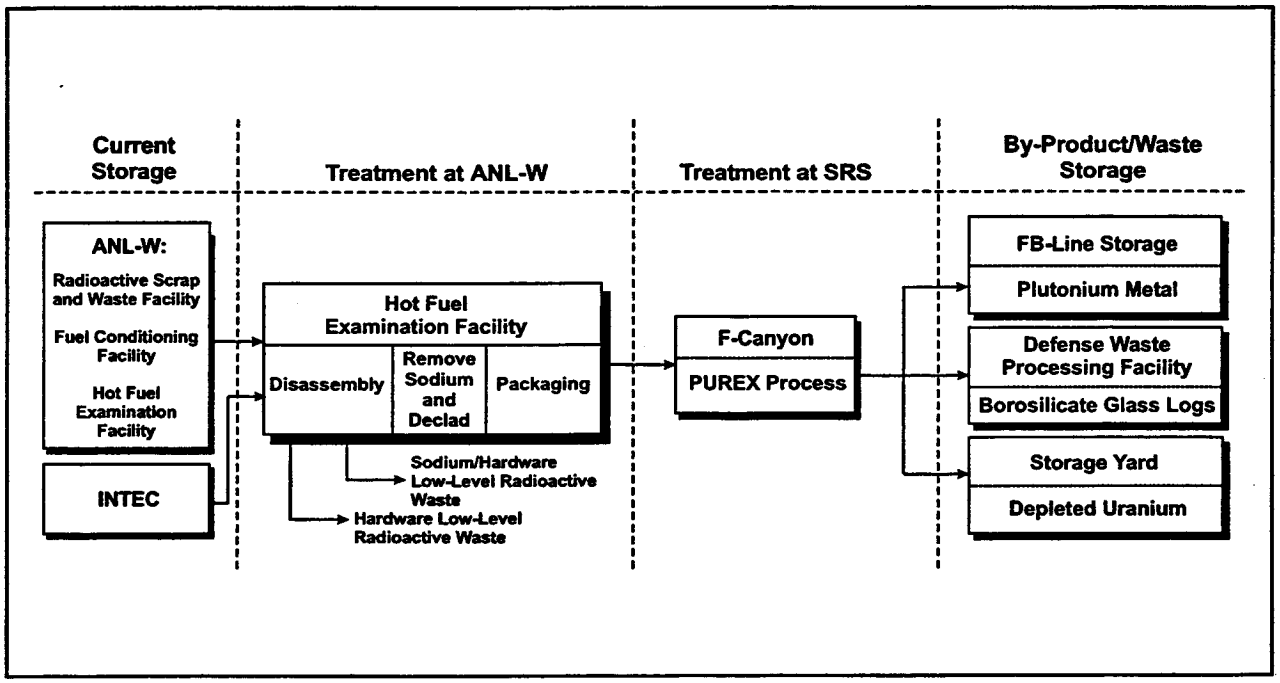

Figure 2–25 Schematic for Blanket Spent Nuclear Fuel Treatment in Alternative 3

2.5.5 Alternative 4: Melt and Dilute Blanket Fuel and Electrometallurgically Treat Driver Fuel at ANL-W

Under this alternative, the blanket spent nuclear fuel elements (approximately 57 metric tons of heavy metal) would be treated at ANL-W using the melt and dilute Option 2 process described in Section 2.3.4. Prior to treatment, the metallic sodium would be removed without decladding at ANL-W, as discussed in Section 2.3.9. Removal of the sodium from the sodium-bonded blanket spent nuclear fuel could take place at the Hot Fuel Examination Facility at ANL-W. Equipment for sodium removal would need to be installed. Equipment necessary for the melt and dilute process would need to be installed at the Hot Fuel Examination Facility, including the addition of the melter and an off-gas system.

Metallic waste resulting from the melt and dilute process containing fission products, depleted uranium, and transuranic elements would be transferred to the Radioactive Scrap and Waste Facility for storage pending ultimate disposal. **Figure 2–26** illustrates the process steps for the sodium-bonded blanket spent nuclear fuel.

Treatment of blanket spent nuclear fuel at ANL-W using the melt and dilute process could start as early as 2005 and could be completed in eight years.

The sodium-bonded driver spent nuclear fuel (approximately 3 metric tons of heavy metal) would be treated at ANL-W using the electrometallurgical treatment process described in Section 2.5.2 for Alternative 1. Treatment of the driver spent nuclear fuel could start as early as 2000 and could be completed in approximately seven years. The process steps for the treatment of the driver sodium-bonded spent nuclear fuel are shown in Figure 2–23.

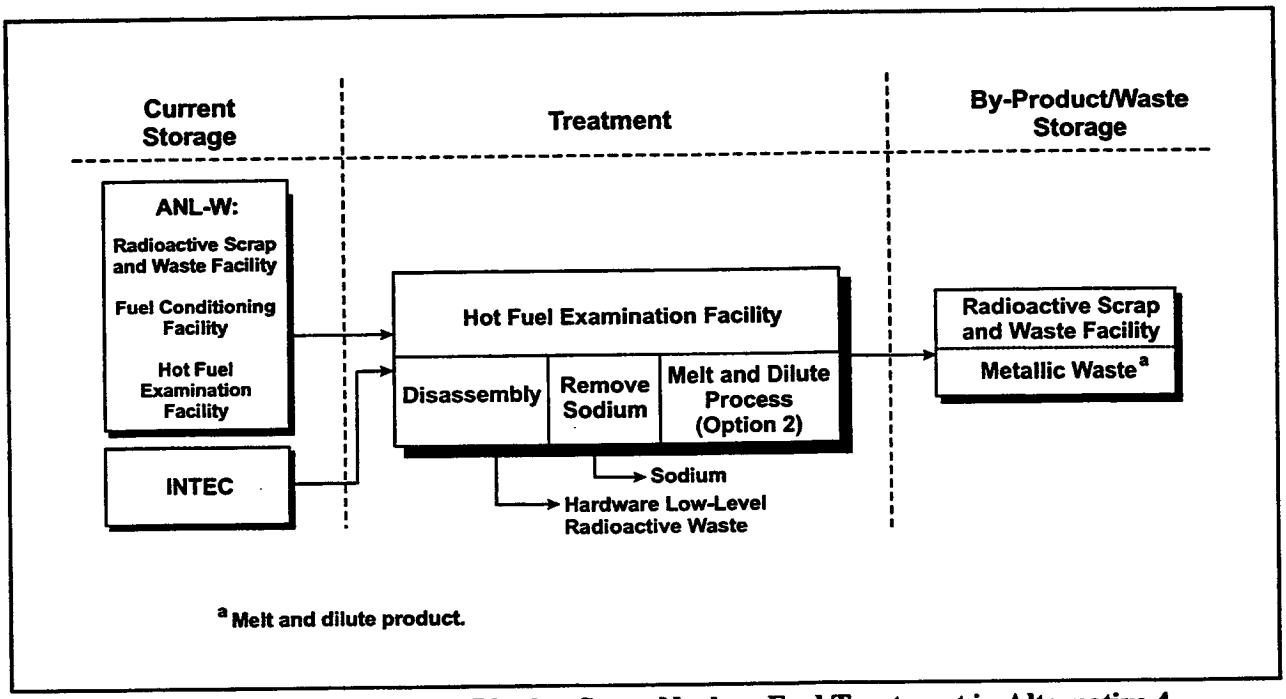

Figure 2–26 Schematic for Blanket Spent Nuclear Fuel Treatment in Alternative 4

2.5.6 Alternative 5: Declad and Clean Blanket Fuel and Electrometallurgically Treat Driver Fuel at ANL-W; Melt and Dilute Blanket Fuel at SRS

Under this alternative, the blanket spent nuclear fuel pins (approximately 57 metric tons of heavy metal) would be packaged and shipped to SRS for treatment. The blanket spent nuclear fuel pins would be separated from the cladding and cleaned to remove the metallic sodium at ANL-W. The declad and cleaned blanket spent nuclear fuel pins would be received at Building 105-L at SRS and treated using the melt and dilute Option 1 process, as described in Section 2.3.4.

Decladding of the sodium-bonded blanket spent nuclear fuel and sodium removal would take place at the Hot Fuel Examination Facility at ANL-W, as discussed in Section 2.3.9. Spent nuclear fuel currently stored at ANL-W facilities could be transported directly to the Hot Fuel Examination Facility. After decladding and sodium removal, the blanket spent nuclear fuel pins would be packaged and stored temporarily at the Hot Fuel Examination Facility pending shipment to SRS.

At SRS, the cans containing the blanket spent nuclear fuel pins would be unpacked at Building 105-L and the blanket spent nuclear fuel pins would be treated using the melt and dilute process. For the purpose of evaluating this alternative, it is assumed that the melt and dilute facility is operational at SRS, as proposed in the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000).

Metallic waste resulting from the melt and dilute process containing fission products, depleted uranium, and transuranic elements would be stored at the L-Area storage pending ultimate disposal.

Figure 2–27 illustrates the process steps for the blanket spent nuclear fuel at ANL-W and SRS.

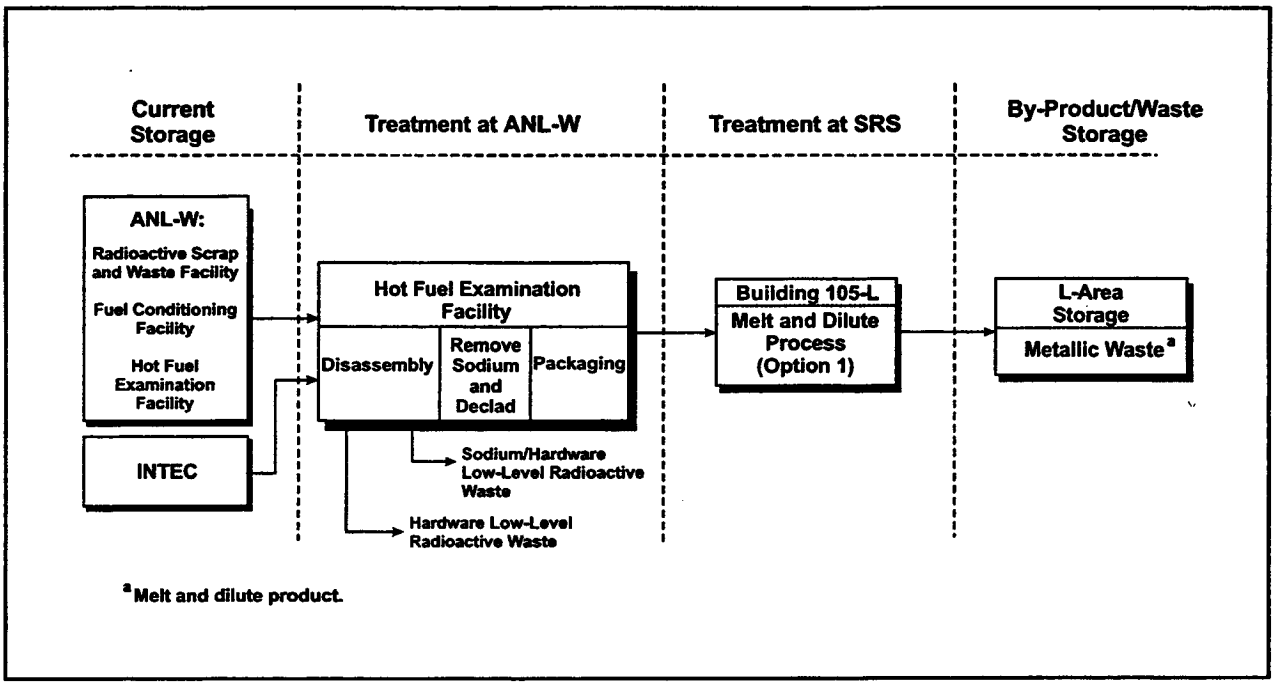

Figure 2–27 Schematic for Blanket Spent Nuclear Fuel Treatment in Alternative 5

Treatment of the blanket spent nuclear fuel at SRS would start around 2035. The facility would be operational in 2005 and is committed to other DOE missions until 2035. If additional capacity becomes available, treatment could start as soon as 2020. The treatment process would last approximately three years. Until 2035, there would be ample time for blanket spent nuclear fuel decladding and sodium removal activities at ANL-W.

The sodium-bonded driver spent nuclear fuel (approximately 3 metric tons of heavy metal) would be treated at ANL-W using the electrometallurgical treatment process described in Section 2.5.2 for Alternative 1. Treatment of the driver spent nuclear fuel at ANL-W could start in 2000 and could be completed in approximately seven years. The process steps for the treatment of the driver sodium-bonded spent nuclear fuel are shown in Figure 2–23.

2.5.7 Alternative 6: Melt and Dilute Blanket and Driver Fuel at ANL-W

Under this alternative, both the sodium-bonded blanket and driver spent nuclear fuel would be treated in the Hot Fuel Examination Facility at ANL-W using Options 2 and 3 of the melt and dilute process discussed in Section 2.3.4. Option 2 would be used for the blanket spent nuclear fuel, and Option 3 would be used for the driver spent nuclear fuel except for 0.1 metric tons (0.11 tons) of oxide, carbide, and nitride fuel, which would not be treated under the alternative. **Figure 2–28** illustrates the steps for the alternative.

Removal of the sodium from the blanket spent nuclear fuel and, to the extent practical, from the driver spent nuclear fuel would take place at the Hot Fuel Examination Facility. Equipment for sodium removal activities and the melt and dilute process would need to be installed in the inert cell of the facility.

The metallic waste resulting from the melt and dilute process containing fission products, depleted uranium, and transuranic elements would be transferred to the Radioactive Scrap and Waste Facility for storage pending ultimate disposal.

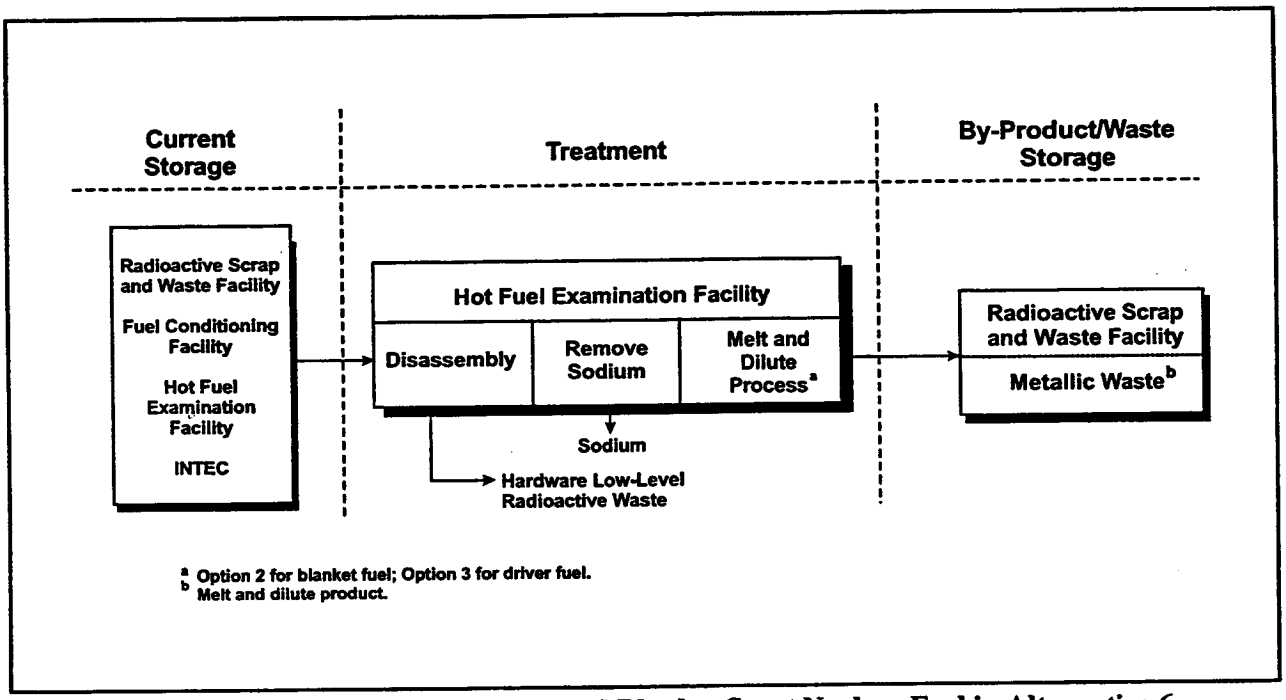

Figure 2–28 Schematic for Driver and Blanket Spent Nuclear Fuel in Alternative 6

The melt and dilute process at ANL-W could start as early as 2003 and would take approximately 12 years to complete for all driver and blanket sodium-bonded spent nuclear fuel.

2.6 Alternatives Considered And Dismissed

In identifying the reasonable alternatives for evaluation in this EIS, two separate issues led to the determination of alternatives that were considered and dismissed: (1) the level of maturity of the alternative technologies; and (2) the level of effort required to modify an existing facility to implement a specific technology. The construction of new facilities when existing facilities are still operative was not considered a reasonable option because of impacts and cost implications. Among the treatment technologies discussed in Section 2.3, the GMODS process and the direct plasma arc-vitreous ceramic process are not as mature as the electrometallurgical, melt and dilute, and PUREX processes when applied to sodium-bonded spent nuclear fuel. The GMODS and plasma arc processes both require significant and extensive research and development before they can be successfully proven to treat sodium-bonded spent nuclear fuel. The GMODS and plasma arc-vitreous ceramic processes each present specific technological challenges that cannot be answered without the construction, operation, and considerable engineering analysis of pilot-scale plants. In comparison, the melt and dilute process is being tested and evaluated, and has been selected as the Preferred Alternative for treatment of almost all aluminum-based spent nuclear fuel at SRS (DOE 2000). Use of the melt and dilute process for sodium-bonded driver spent nuclear fuel only requires technology enhancement, which DOE already has proposed for treating other spent nuclear fuel. In addition, unlike the other technologies that would require no new construction, GMODS and plasma arc processes would require the installation of large, specialized equipment in new hot cell facilities, the size and complexity of which are not sufficiently determined to allow detailed environmental impact analysis.

GMODS Process

The GMODS process, although similar to the melt and dilute process because of its thermal treatment, has not been developed beyond the laboratory scale. Several developmental steps would be required before it could be deemed a mature process. These include: detailed process development, resolution of containment concerns, testing, and a pilot plant demonstration to address technology risks (e.g., reliability and throughput).

GMODS would require large, specialized equipment to be installed in eight new large hot cell facilities. GMODS would dissolve the fuel elements or fuel assemblies entirely in a lead/lead-oxide system. An off-gas treatment system similar to that for the melt and dilute process would be required to treat the radioactive elements volatilized at about 1,000 °C (1,830 °F). The GMODS equipment could produce an intermediate waste form containing most of the actinides, fission products, and structural materials. After some preprocessing, the waste stream would be fed into the melter for the production of a new type of borosilicate glass log. These logs would contain uranium, other actinides, and structural elements in addition to the fission products. Because of the highly corrosive nature of the chemicals in the system, the technical feasibility of the alternative has not been established. This would add an additional degree of uncertainty to the waste estimates, as well as to the ultimate success of the fuel conditioning project.

Direct Plasma Arc-Vitreous Ceramic Process

The direct plasma arc-vitreous ceramic process is being used for the vitrification of low-level mixed waste. However, vitrification of spent nuclear fuel by this process is understood only on a conceptual level. The plasma arc treatment method would require large, complex equipment to be installed in a new, specially constructed hot cell facility. Such a facility could be constructed next to the Hot Fuel Examination Facility at ANL-W to secure some services. It would require the installation of equipment to cut the fuel assemblies into small pieces, a ceramic melter (furnace) to melt and oxidize the pieces at temperatures at least as high as 1,600 °C (2,900 °F), and an off-gas treatment system. As with the GMODS and melt and dilute processes, uranium and plutonium are not separated during the process. The conditioned spent nuclear fuel form would be vitreous ceramic and would include the sodium in a stable form. As with all processes that dissolve or melt spent nuclear fuel, the plasma arc process would produce radioactive off-gases. These gases would be filtered and treated, and the filter and treatment media would be stabilized into an acceptable waste form by a yet-to-be-determined process. The process would require testing in a pilot-scale plant to address the reliability of the plasma system.

The high temperatures of the process could increase the radioactive materials available for release during normal operation and accident conditions, thus increasing the exposure risk to members of the general public. Compared to other alternatives, there is a substantial uncertainty about the risk from accident conditions, considering the complexity of the off-gas treatment system. Because of the high temperature, more radioactive elements would be volatilized. In addition, considerable development would be required to produce very high-temperature rotating equipment that would operate reliably in a hot cell environment.

Chloride Volatility Process

The chloride volatility process design is in an early conceptual stage. The process needs high temperatures and chlorination for volatilization and chemical reactions to separate various fission products from uranium. This treatment technology would require a very elaborate gaseous separation process with potentially significant occupational and public risk in comparison to other treatment technologies, from both the volatilized fission products and the chlorine gas.

Electrometallurgical Treatment at INEEL Test Area North

Treatment of sodium-bonded spent nuclear fuel using the electrometallurgical treatment process at INEEL's Test Area North was considered and dismissed because Test Area North would require extensive modification to treat sodium-bonded spent nuclear fuel. Implementation of this alternative would require the construction of an argon hot cell. In addition, it would require either the procurement of new equipment or the transfer of already-contaminated equipment and other systems existing at ANL-W.

Treatment of Driver or Clad Blanket Spent Nuclear Fuel Using SRS PUREX Process

As discussed in Section 2.3.2, use of the PUREX process facilities at SRS for the treatment of sodium-bonded spent nuclear fuel would require the development and installation of a versatile front-end process to handle mechanical decladding, sodium removal, and zirconium sludge formation for EBR-II spent nuclear fuel. Such development does not appear justified for the sole purpose of treating the relatively small quantity of driver spent nuclear fuel.

Treatment of Driver Spent Nuclear Fuel Using SRS Melt and Dilute Process

As discussed in Section 2.3.4, the treatment of driver spent nuclear fuel would require a modified melt and dilute process that would be capable of handling the sodium volatilized from processing chopped driver spent nuclear fuel elements with the cladding intact. To accomplish this at SRS, significant design changes would be required from the process that DOE has proposed for the aluminum-clad spent nuclear fuel, which does not contain sodium. These design changes do not appear justified for the sole purpose of treating the relatively small quantity of driver spent nuclear fuel.

Treatment Using INEEL PUREX Process

Sodium-bonded spent nuclear fuel from EBR-II was being processed at the Idaho Chemical Processing Plant (now INTEC) using a PUREX process. DOE stopped processing at INTEC as a matter of policy in 1992, and the facility was permanently shut down. Reactivation of the facility is not practical and the alternative was dismissed.

2.7 Ultimate Disposition

One of the technical risks in implementing any of the sodium-bonded spent nuclear fuel treatment methods is the uncertainty surrounding the acceptability of DOE spent nuclear fuel for placement in a potential repository. DOE would receive a license from the NRC to receive and store spent nuclear fuel in a repository (10 CFR 60 or draft 10 CFR 63). In order to obtain a license, DOE must develop acceptance criteria that establish the condition of the spent nuclear fuel for disposal and demonstrate that the criteria will meet NRC standards. Any spent nuclear fuel packaging or treatment technology must be capable of putting fuel in a form that will satisfy the acceptance criteria requirements. DOE's Office of Civilian Radioactive Waste Management has responsibility for a Federal repository. It is working to refine its acceptance criteria to ensure that spent nuclear fuel and high-level radioactive waste are packaged suitably for disposal. DOE has drafted preliminary acceptance criteria that are being used to assess the feasibility of DOE spent nuclear fuel disposition options (DOE 1999a). If the repository is developed, final acceptance criteria will not be available until after the NRC issues its construction authorization, based on the successful demonstration of safe, long-term performance of the candidate repository in accordance with NRC regulations. Until such time, the preliminary acceptance criteria tend to be conservative to allow for uncertainties in performance of engineered and natural barriers and how such performance will impact public and worker health and safety, and material isolation.

To ensure that the treatment option DOE selects will result in a product that is likely to meet the acceptance criteria, DOE is working with the NRC to obtain comments on the research and development work that DOE will perform to establish treatment technology specifications. To provide additional independent evaluation of the suitability of new treatment technologies, DOE requested that the National Academy of Sciences' National Research Council provide recommendations regarding DOE's sodium-bonded spent nuclear fuel treatment and disposition program. In a recent report (NAS 1998), the National Research Council recommended that the Electrometallurgical Treatment Research and Demonstration Project be carried to completion. The Council also expressed the opinion that, with the exception of the PUREX process, all other alternatives to the electrometallurgical process were at an early stage of development.

2.8 Preferred Alternative

Council on Environmental Quality regulations (40 CFR 1502.14e) require that an agency identify its preferred alternative(s) in the final EIS. After careful consideration of public comments and programmatic, environmental, nonproliferation, and cost issues, DOE has identified electrometallurgical treatment as its Preferred Alternative for the treatment and management of all sodium-bonded spent nuclear fuel, except for the Fermi-1 blanket fuel. The No Action Alternative is preferred for the Fermi-1 blanket spent nuclear fuel. Thus, the Preferred Alternative is a combination of Alternative 1 and the No Action Alternative. This combination would result in 26 metric tons of heavy metal of EBR-II and miscellaneous spent nuclear fuel being treated using the electrometallurgical process and 34 metric tons of heavy metal of Fermi-1 blanket spent nuclear fuel remaining in storage, pending a subsequent decision on its long-term management. The environmental consequences of the Preferred Alternative are addressed in Section 4.10.

DOE will validate the cost of using alternative treatment techniques (e.g., sodium removal and placement in high-integrity cans) for the Fermi-1 blanket spent nuclear fuel. These techniques may be economically favorable for the Fermi-1 blanket spent nuclear fuel because of characteristics that distinguish it from the EBR-II spent nuclear fuel. For example, the Fermi-1 blanket spent nuclear fuel does not require the extensive safeguards and security measures that are required for the EBR-II blanket fuel. The difference in security requirements for these two types of fuel is a result of the difference in plutonium content. The EBR-II blanket fuel has 30 times more plutonium at a greater concentration than the Fermi-1 blanket fuel.

Should DOE select the Preferred Alternative in the Record of Decision, DOE would proceed with the electrometallurgical treatment of the EBR-II sodium-bonded spent nuclear fuel and monitor the results and costs while continuing the development of sodium removal techniques for the Fermi-1 blanket spent nuclear fuel. Sodium removal would increase the number of long-term management options for the Fermi-1 fuel. While EBR-II spent nuclear fuel is undergoing electrometallurgical treatment and the Fermi-1 spent nuclear fuel remains in storage, DOE has approximately four years in which to evaluate the operating experience of electrometallurgical treatment technology and further develop other alternatives for the Fermi-1 spent nuclear fuel. After these data are evaluated, DOE would decide whether to treat the Fermi-1 blanket spent nuclear fuel using electrometallurgical treatment or to use another treatment method and/or disposal technique.

Before making a decision to treat or dispose of the Fermi-1 blanket spent nuclear fuel, DOE will determine whether the analysis in this EIS is adequate to support a subsequent Record of Decision or whether additional NEPA review is required. In any case, DOE will notify the public of its preferred approach for the Fermi-1 blanket spent nuclear fuel at least 30 days before issuing a Record of Decision regarding treatment or disposal.

For several years, DOE has been actively developing electrometallurgical treatment technology specifically for the management of sodium-bonded spent nuclear fuel. Having completed a successful demonstration of electrometallurgical treatment, DOE believes that this technology has the highest probability of meeting the Department's needs for managing much of the sodium-bonded spent nuclear fuel. Electrometallurgical technology would convert the reactive fuel into ceramic and metallic waste forms, both of which are more

stable than untreated sodium-bonded spent nuclear fuel. In addition, uranium would be separated from the spent nuclear fuel, blended with depleted uranium if needed to lower enrichment levels, and cast into ingots to be stored until a disposition decision is made through a separate NEPA review. Most of the plutonium would be disposed of in the ceramic waste form, with the remaining small fraction disposed of in the metallic waste form. Currently, the only waste form that has been tested and analyzed extensively under geologic repository conditions and may be accepted for repository disposal is borosilicate glass. Tests have shown the ceramic and metallic waste forms from electrometallurgical treatment may perform as well as the standard borosilicate glass waste form. The ceramic and metallic waste forms would require less storage volume than untreated spent nuclear fuel.

2.9 Summary of Environmental Impacts

This section summarizes the environmental impacts associated with the No Action Alternative and the six reasonable alternatives under the proposed action that are evaluated in detail as part of this EIS (see Section 2.5). The information presented in this section is based on Chapter 4, which provides a detailed discussion of the impacts on potentially affected environmental areas. Such environmental areas include: air quality, water resources, socioeconomics, public and occupational health and safety, environmental justice, waste management, and transportation.

For the alternatives evaluated, the analyses showed that there would be no significant impacts on air quality, water resources, socioeconomics, public and occupational health and safety, environmental justice, and transportation. The radiological and nonradiological gas and liquid releases, as well as the associated exposures to workers and the public, would be well below regulatory standards and guidelines and no mitigation measures would be warranted. In addition, the environmental impact analysis indicates that there are no significant impacts that would discriminate one alternative over another.

The only significant difference between the No Action Alternative and the reasonable alternatives is in the area of waste generation. All of the proposed alternatives result in a decrease in high-level radioactive waste volume as compared to the direct disposal volume associated with the No Action Alternative.

For the No Action Alternative and the six alternatives evaluated, the proposed facilities already exist. Except for internal building modifications and new equipment installation, no construction activities would be required. Therefore, DOE has determined the proposed action would have minimal or no impacts on land resources, visual resources, noise, geology and soils, ecological resources, and cultural and paleontological resources. Impacts to these resources were not evaluated in detail in the EIS.

The impacts of the No Action Alternative are presented first as a baseline for comparing the impacts under the proposed action. A summary of the environmental impacts for the No Action Alternative and the other six reasonable alternatives is presented as **Table 2–4**. Section 4.1, Chapter 4, provides information on the categories and results illustrated in Table 2–4.

2.9.1 No Action Alternative Impacts

Under the No Action Alternative, the sodium-bonded spent nuclear fuel would not be treated (no sodium would be removed from the interior of the fuel elements). The EIS evaluates the impacts of two separate options under this alternative:

a. Monitoring and stabilizing the sodium-bonded spent nuclear fuel as necessary for continued storage at current locations until 2035, or until a new treatment technology (such as GMODS or plasma arc) is developed.

b. Direct disposal of sodium-bonded spent nuclear fuel in a geologic repository by packaging the fuel in high-integrity cans. As discussed in Section 2.3.8, the direct disposal option may not meet current NRC and/or RCRA requirements.

The activities associated with the preparation of sodium-bonded spent nuclear fuel for direct disposal would be similar to those needed to prepare the fuel for interim or continued storage. Both require that fuel be transferred to a hot cell, examined (nondestructive examination) and characterized, and repackaged. The only difference between these two options is that for direct disposal, the sodium-bonded spent nuclear fuel would be placed in high-integrity cans in preparation for ultimate disposal, while for storage it would not be placed in high-integrity cans. Direct disposal also requires consideration of criticality safety, thereby limiting the amount of driver spent nuclear fuel that could be packaged in a canister, leading to higher repository volume requirements.

Air Quality

For both options under the No Action Alternative, activities at ANL-W and INTEC would have a negligible impact on existing air quality. Radiological emissions would also be low and well below regulatory concern. Approximately 810 curies would be released over a 35-year period from possible fuel degradation during storage. Air quality for INEEL is not expected to change as a result of the No Action Alternative.

Water Resources

Surface water is not used at ANL-W and INTEC and this would not change under either option of the No Action Alternative. Groundwater use, primarily domestic consumption, could decrease if there is a reduction in workers at ANL-W.

No changes are expected in liquid effluent discharges. There are currently no discharges to surface waters (radiological or nonradiological) except for discharges of nonhazardous liquid waste, which are monitored and subject to National Pollutant Discharge Elimination System (NPDES) permit requirements.

Socioeconomics

Under either option of the No Action Alternative, there could be a reduction of approximately 350 workers at ANL-W. This reduction could result in the loss of 940 additional indirect jobs in the economic region. The reduction would take place over time, therefore, the No Action Alternative would not result in any noticeable changes in the existing regional economy, housing characteristics, or community services.

Public and Occupational Health and Safety

The risk to the health and safety of workers and the public under either option of the No Action Alternative would be from the potential exposure to radiological or hazardous chemical emissions during normal operation or accident conditions.

Radiological Exposures

Routine radioactive releases associated with either option of the No Action Alternative at ANL-W and INTEC would be small. The maximum annual dose to the population within 80 kilometers (50 miles) from these releases would be 0.0015 person-rem. This exposure would lead to 7.5×10^{-7} additional latent cancer fatalities in the population, or one chance in 1.3 million that the exposed population would experience a latent cancer fatality. For comparison purposes, the collective dose for the same population in the year 2010 from natural background radiation would be 86,250 person-rem. The maximally exposed offsite individual would receive

a maximum of 0.00026 millirem, and the risk of developing a fatal cancer from this exposure would be 1.3×10^{-10}, or one chance in 7.7 billion. The average worker would receive 60 millirem per year, and the risk of developing a cancer from this exposure would be 0.000024, or one chance in 41,666.

Similarly, for the 35-year duration of the project, the population dose from routine releases would be 0.013 person-rem (with a latent cancer fatality risk of 6.5×10^{-6}); the maximally exposed offsite individual dose would be 0.0023 millirem (with a latent cancer fatality risk of 1.1×10^{-9}); and the worker population dose would be 209 person-rem (with a latent cancer fatality risk of 0.084).

The maximum annual cancer risk from postulated accident conditions under continued storage or direct disposal of the No Action Alternative at ANL-W would be 5.6×10^{-6} for the population within 80 kilometers (50 miles). The annual cancer risk for the maximally exposed offsite individual would be 4.8×10^{-8}, and 1.5×10^{-8} for the noninvolved worker.

Hazardous Chemical Exposures

Hazardous chemical impacts resulting from either option of the No Action Alternative would be small because any emissions of hazardous chemicals from activities under the No Action Alternative would be very low.

Hazardous chemical impacts under accident conditions, evaluated in terms of Emergency Response Planning Guideline (ERPG) values, indicate that under either option of the No Action Alternative, the worst postulated accident conditions would result in less than ERPG-1 conditions for a worker or the maximally exposed offsite individual.

Environmental Justice

As discussed above, the impacts from either option of the No Action Alternative on the health and safety of the public would be very small regardless of the racial and ethnic composition of the population and independent of the economic status of the individuals comprising the population in 2010.

Waste Management

For both options under the No Action Alternative, various types of waste would continue to be generated at ANL-W. These include low-level radioactive, transuranic, mixed, hazardous, and nonhazardous waste. These waste types are associated with the operation of the facilities where the sodium-bonded spent nuclear fuel is stored. High-level radioactive waste in metallic and ceramic forms generated as a result of completing the Electrometallurgical Treatment Research and Demonstration Project waste processing would be stored at the Radioactive Scrap and Waste Facility pending disposal. Finally, some additional low-level radioactive waste and transuranic waste would be generated from the deactivation of the demonstration project. The volumes of these waste types are presented in Table 2–4.

Transportation

The No Action Alternative involves the movement of sodium-bonded spent nuclear fuel within the INEEL site. All fuel stored at INTEC would be moved to ANL-W for repackaging in preparation for transport either to a geologic repository or out of the State of Idaho by 2035.

The dose to transportation workers from all transportation activities under the No Action Alternative is estimated at 0.003 person-rem; the dose to the public would be 0.022 person-rem. Accordingly, incident-free transportation of radioactive material would result in 1.2×10^{-6} latent cancer fatalities among transportation

workers and 0.000011 latent cancer fatalities in the total affected population over the duration of the transportation activities.

The dose to the affected population from postulated accidents during transportation would be less than 1.0×10^{-6} person-rem, resulting in less than 1.0×10^{-9} latent cancer fatalities. Nonradiological transportation accidents are estimated to result in 0.00012 traffic fatalities.

2.9.2 Proposed Action Impacts

Under the proposed action, the EIS evaluates six distinct alternatives, as described in Section 2.5 and illustrated in Figure 2-22. Alternative 1 proposes to treat both driver and blanket spent nuclear fuel using the electrometallurgical method at ANL-W. Alternatives 2 through 5 propose to treat the driver spent nuclear fuel using the electrometallurgical method (as in Alternative 1), but other methods and/or sites would be used for the blanket spent nuclear fuel, including: the high-integrity can packaging at ANL-W (Alternative 2); the PUREX process at SRS (Alternative 3); the melt and dilute process at ANL-W (Alternative 4); and the melt and dilute process at SRS (Alternative 5). Alternative 6 proposes to treat both driver and blanket spent nuclear fuel using the melt and dilute method at ANL-W.

All alternatives under the proposed action would have very small impacts on air quality, water resources, socioeconomics, public and occupational health and safety, environmental justice, and transportation areas of the environment in and around the INEEL/ANL-W and SRS locations. For all alternatives, the radiological and nonradiological gaseous emissions and liquid effluent, as well as the associated exposures to workers and the public, would be well below regulatory standards and guidelines. A major difference between the No Action and proposed action alternatives is in the area of waste generation. Since the acceptability of chemically reactive sodium in a high-level radioactive waste repository is a primary concern in this EIS, it is important to consider the volume of high-level radioactive waste for all the proposed action alternatives. All the proposed action alternatives would result in a decrease in high-level radioactive waste volume as compared to the direct disposal No Action Alternative. The reduction in high-level radioactive waste volume for each alternative would be: 47 percent (Alternative 1); 71 percent (Alternative 2); 84 percent (Alternative 3); 58 percent (Alternative 4); 37 percent (Alternative 5); and 43 percent (Alternative 6).

Air Quality

The proposed action would have a negligible impact on existing air quality at ANL-W and SRS for each of the alternatives. Air quality at ANL-W and SRS would not change as a result of the proposed action.

During the year of maximum releases, radiological gaseous emissions would be well below regulatory concerns for each of the alternatives. Radiological gaseous emissions at ANL-W would be in the range of 770 (Alternative 1) to 2,162 (Alternative 6) curies of elemental tritium and 11,600 (Alternative 1) to 32,650 (Alternative 6) curies of krypton-85.

During the year of maximum releases, radiological gaseous emissions at SRS would be 54 (Alternative 5) to 162 (Alternative 3) curies of elemental tritium and 396 (Alternative 5) to 1,187 (Alternative 3) curies of krypton-85.

Water Resources

Surface water is not used at ANL-W, and this would not change under any of the alternatives proposed for ANL-W. Groundwater use, primarily domestic consumption, would remain at current levels, as the work force would be expected to remain at current levels for all alternatives of the proposed action.

No changes would be expected in liquid effluent discharges from any of the alternatives at ANL-W. There are currently no discharges to surface waters (radiological or nonradiological) except for discharges of nonhazardous liquid waste to the industrial pond, which are monitored and are subject to NPDES permit requirements.

Potential radioactive liquid effluent has been identified for the PUREX process at SRS under Alternative 3. Table 2–4 indicates some small quantities of tritium, and other isotopes. No radioactive liquid effluent has been identified for the melt and dilute process at SRS under Alternative 5.

Socioeconomics

All the alternatives under the proposed action assume that the treatment and management of the sodium-bonded spent nuclear fuel at ANL-W or SRS would not require an additional work force, but the activities would keep the work force from being reduced. Therefore, there would be no changes to the socioeconomic conditions in the vicinity of either ANL-W or SRS.

Public and Occupational Health and Safety

The potential risk of concern to the health and safety of the workers and the public under the proposed action would be from exposure to routine radiological emissions and hazardous chemical releases under normal operation or accident conditions. As indicated in Table 2–4, the risk would be small for all alternatives considered under the proposed action.

Radiological Exposures

Comparing alternatives at ANL-W, the maximum annual population dose from routine gaseous radioactive releases would range from 0.0028 person-rem (Alternative 1) to 0.012 person-rem (Alternative 6), with latent cancer fatalities in the range of 1.4×10^{-6} to 6.0×10^{-6}, respectively. The project total population dose would range from 0.016 person-rem (Alternative 1) to 0.024 person-rem (Alternative 6), with latent cancer fatalities in the range of 8.2×10^{-6} to 1.2×10^{-5}, respectively.

The maximum annual dose to the maximally exposed offsite individual at ANL-W would range from 0.00034 millirem (Alternative 1) to 0.002 millirem (Alternative 6), with a latent cancer fatality risk in the range of 1.7×10^{-10} to 1.0×10^{-9}, respectively. The project total dose to the maximally exposed offsite individual would range from 0.002 millirem (Alternative 1) to 0.004 (Alternative 6), with latent cancer fatality risks of 1.0×10^{-9} to 2.0×10^{-9}, respectively.

The collective annual dose to workers at ANL-W would be 22 person-rem for all alternatives. This corresponds to additional latent cancer fatalities of 0.0088. The average dose to a worker at ANL-W would be 60 millirem per year, which corresponds to a latent cancer fatality risk of 0.000024 per year. The project total worker dose would range from 231 person-rem (Alternatives 3 and 5) to 319 person-rem (Alternatives 1 and 4), with latent cancer fatalities ranging between 0.092 and 0.13, respectively.

Comparing alternatives at SRS, the maximum population dose from routine gaseous radioactive releases would range from 0.0076 person-rem per year for three years (Alternative 5) to 0.02 person-rem for the whole treatment period (Alternative 3), corresponding to additional latent cancer fatalities in the range of 0.000011 to 0.00001, respectively, for the whole treatment period.

The maximum dose to the maximally exposed offsite individual would range from 0.0001 millirem per year for three years (Alternative 5) to 0.00051 millirem (Alternative 3) for the whole treatment period, with latent cancer fatality risks of 1.5×10^{-10} and 2.6×10^{-10}, respectively, for the whole treatment period.

The maximum collective dose to workers at SRS would be 50 person-rem per year (Alternative 5) for three years. This corresponds to 0.06 additional latent cancer fatalities. The maximum average dose to a worker at SRS would be 500 millirem per year (Alternative 5), which corresponds to a latent cancer fatality risk of 0.0002 per year.

The highest annual latent cancer fatality risk for the population within 80 kilometers (50 miles) of ANL-W from postulated design-basis accident conditions under the proposed action would be 0.009 (Alternative 6, driver fuel, design-basis earthquake). The highest annual latent cancer fatality risk for the maximally exposed offsite individual would be 0.000076 (Alternative 6, driver fuel, design-basis earthquake). The highest annual latent cancer fatality risk for the noninvolved worker would be 2.7×10^{-6} (Alternative 6, driver fuel, design-basis earthquake).

The highest annual latent cancer fatality risk for the population within 80 kilometers (50 miles) of ANL-W from postulated beyond-design-basis accident conditions under the proposed action would be 0.000013 (Alternatives 1 through 5, driver fuel, beyond-design-basis earthquake). The highest annual latent cancer fatality risk for the maximally exposed offsite individual would be 2.2×10^{-7} (Alternatives 1 through 5, driver fuel, beyond-design-basis earthquake). The highest annual latent cancer fatality risk for the noninvolved worker would be 2.3×10^{-9} (Alternatives 1 through 5, blanket fuel, beyond-design-basis earthquake).

The highest annual latent cancer fatality risk for the population within 80 kilometers (50 miles) of SRS from postulated design-basis accident conditions under the proposed action would be 0.011 (Alternative 5, blanket fuel, loss of power). The highest annual latent cancer fatality risk for the maximally exposed offsite individual would be 6.6×10^{-6} (Alternative 5, blanket fuel, loss of power). The highest annual latent cancer fatality risk for the noninvolved worker would be 3.4×10^{-7} (Alternative 5, blanket fuel, loss of power).

Hazardous Chemical Exposures

Hazardous chemical impacts from normal operations for all alternatives under the proposed action would be small because the emissions of hazardous chemicals from the treatment and management of sodium-bonded spent nuclear fuel would be very low.

Hazardous chemical impacts under accident conditions, evaluated in terms of comparison to ERPG values, indicate that under the proposed action, all postulated hazardous chemical releases would not result in worse than ERPG-1 conditions for the noninvolved worker or the maximally exposed offsite individual at either ANL-W or SRS.

Environmental Justice

As discussed above, the impacts from the proposed action on the health and safety of the public would be very small, regardless of the racial and ethnic composition of the population and independent of the economic status of the individuals comprising the population in 2010.

Waste Management

Table 2–4 presents a comparison of the volumes of high-level radioactive, low-level radioactive, and transuranic waste generated by each of the alternatives. The alternatives would generate from 37 to 84 percent less high-level radioactive waste as compared to the direct disposal option of the No Action Alternative. Each of the alternatives would generate more transuranic waste, but Alternatives 1, 2, 4, and 6 would exceed this waste volume by a range of only 7 to 41 percent. Alternatives 3 and 5 would generate significantly greater volumes of transuranic waste, between 2.7 to 10 times the volume of transuranic waste generated by the direct disposal No Action Alternative.

All of the alternatives either would remove or convert the metallic sodium into a nonreactive form.

With respect to disposability and waste acceptance criteria, only the borosilicate glass waste form of Alternative 3 for blanket spent nuclear fuel has been tested and analyzed extensively under conditions relevant to a geologic repository. It is expected that other waste forms (e.g., ceramic, metallic, and possibly high-integrity cans not containing metallic sodium) would be suitable for repository disposal.

Transportation

The transportation activities for all alternatives under the proposed action would involve the movement of the sodium-bonded spent nuclear fuel within the INEEL site.

The incident-free dose to transportation workers from these activities would be between 0.0043 (Alternative 2) and 0.027 (Alternative 6) person-rem; the dose to the affected public would be between 0.032 (Alternative 2) and 0.2 (Alternative 6) person-rem. Accordingly, incident-free transportation activities would result in latent cancer fatalities in the range of 1.7×10^{-6} to 1.1×10^{-5} among transportation workers and 0.000016 to 0.0001 among the total affected population over the duration of the transportation activities.

The dose to the affected population from postulated accidents from these activities would be less than 1×10^{-6} person-rem, resulting in less than 1×10^{-9} latent cancer fatalities. Nonradiological traffic fatalities would be approximately 0.0001.

Transportation activities under Alternatives 3 and 5 would include, in addition, the movement of the blanket spent nuclear fuel pins from ANL-W to SRS. The incident-free dose to transportation workers from these activities would be 0.0012 person-rem; the dose to the public would be 0.012 person-rem. Accordingly, incident-free transportation activities would result in 4.7×10^{-7} latent cancer fatalities among transportation workers and 6.0×10^{-6} latent cancer fatalities in the total affected population over the duration of the transportation activities. Nonradiological fatalities from vehicle emissions during intersite transportation would be 0.00039 among affected urban populations along the transportation route.

The dose to the affected population from postulated accidents from these activities would be less than 3.0×10^{-6} person-rem, resulting in less than 1.5×10^{-9} additional latent cancer fatalities. Nonradiological traffic fatalities would be 0.0018.

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Table 2–4 Summary of Environmental Consequences for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel

Resource/Material Categories	*No Action* ANL-W	*Alternative 1* ANL-W	*Alternative 2* ANL-W
Air Quality - Radiological air emissions (curies per year)	Negligible impact 811 [a]	Negligible impact Tritium: 770 Krypton-85: 11,600	Negligible impact Tritium: 809 Krypton-85: 11,860
Water Resources - Radiological liquid effluent (curies per year)	No impact No liquid effluent	No impact No liquid effluent	No impact No liquid effluent
Socioeconomics	Loss of 350 direct jobs and 940 indirect jobs. No noticeable impact.	Work force maintained; No impact	Work force maintained; No impact

Public and Occupational Health and Safety						
• Project duration	35 years		13 years		9 years	
• Normal operations (annual) [c]	**Dose per year**	**LCF**	**Dose per year**	**LCF**	**Dose per year**	**LCF**
- Population	0.0015	7.5×10^{-7}	0.0028	1.4×10^{-6}	0.003	1.5×10^{-6}
- MEI	0.00026	1.3×10^{-10}	0.00034	1.7×10^{-10}	0.00037	1.9×10^{-10}
- Average individual	6.2×10^{-6}	3.1×10^{-12}	0.000012	5.8×10^{-12}	0.000013	6.2×10^{-12}
- Worker population	22	0.0088	22	0.0088	22	0.0088
- Average worker	60	0.000024	60	0.000024	60	0.000024
• Normal operations (project total) [c]	**Dose**	**LCF**	**Dose**	**LCF**	**Dose**	**LCF**
- Population	0.013	6.5×10^{-6}	0.016	8.2×10^{-6}	0.017	8.3×10^{-6}
- MEI	0.0023	1.1×10^{-9}	0.002	1.0×10^{-9}	0.0021	1.0×10^{-9}
- Worker population	209	0.084	319	0.13	231	0.092
Hazardous chemicals						
- MEI	None		None		None	
• Accidents						
Maximum annual cancer risk						
- Population	5.6×10^{-6} (DBA) [a]		5.6×10^{-6} (DBA); 0.000013 (BDBA)		5.6×10^{-6} (DBA); 0.000013 (BDBA)	
- MEI	4.8×10^{-8} (DBA) [a]		4.8×10^{-8} (DBA); 2.2×10^{-7} (BDBA)		4.8×10^{-8} (DBA); 2.2×10^{-7} (BDBA)	
- Noninvolved worker	1.5×10^{-8} (DBA) [a]		4.5×10^{-8} (DBA); 2.3×10^{-9} (BDBA)		1.5×10^{-8} (DBA); 1.5×10^{-9} (BDBA)	
Chemical accidents						
- MEI	Less than ERPG-1		Less than ERPG-1		Less than ERPG-1	
- Noninvolved worker	Less than ERPG-1		Less than ERPG-1		Less than ERPG-1	
Environmental Justice	No disproportionately high and adverse impact to minority or low-income populations					
Waste Management (cubic meters)						
• High-level radioactive waste	152 (Direct disposal volume) [d]		81.1		43.2 [e]	
• Low-level radioactive waste	904		862		733.7	
• Transuranic waste	12.1		14.1		10.7	
Transportation						
• Incident-free	**Person-rem**	**LCF**	**Person-rem**	**LCF**	**Person-rem**	**LCF**
- Population	0.022	0.000011	0.033	0.000016	0.032	0.000016
- Workers	0.003	1.2×10^{-6}	0.0044	1.8×10^{-6}	0.0043	1.7×10^{-6}
• Accidents						
- Population	less than 1.0×10^{-6}	less than 1.0×10^{-9}	less than 1.0×10^{-6}	less than 1.0×10^{-9}	less than 1.0×10^{-6}	less than 1.0×10^{-9}

ERPG = Emergency Response Planning Guideline; LCF = Latent Cancer Fatalities; MEI = Maximally Exposed Offsite Individual;
DBA = Design-Basis Accident; BDBA = Beyond-Design-Basis Accident

[a] Represents total curies for 35 years; tritium: 51 curies; krypton-85: 760 curies; iodine-129: 0.000018 curies.

[b] Over a period of six months.

Alternative 3		Alternative 4	Alternative 5		Alternative 6
ANL-W	SRS [b]	ANL-W	ANL-W	SRS	ANL-W
Negligible impact Tritium: 809 Krypton-85: 11,860	Negligible impact Tritium: 162 Krypton-85: 1,187	Negligible impact Tritium: 809 Krypton-85: 11,860	Negligible impact Tritium: 809 Krypton-85: 11,860	Negligible impact Tritium: 54 Krypton-85: 396	Negligible impact Tritium: 2,162 Krypton-85: 32,650
No impact No liquid effluent	Negligible impact Tritium: 1.54 Other: less than 0.022	No impact No liquid effluent	No impact No liquid effluent	No impact No liquid effluent	No impact No liquid effluent
Work force maintained; no impact	Work force maintained; no impact	Work force maintained; no impact	Work force maintained; no impact	Work force maintained; no impact	Work force maintained; no impact
9 years	Less than 1 year	13 years	9 years	3 years	12 years
Dose per year / LCF	Dose per year / LCF	Dose per year / LCF	Dose per year / LCF	Dose per year / LCF	Dose per year / LCF
0.003 / 1.5×10^{-6}	0.02 / 0.000010	0.003 / 1.5×10^{-6}	0.003 / 1.5×10^{-6}	0.0076 / 3.8×10^{-6}	0.012 / 6.1×10^{-6}
0.00037 / 1.9×10^{-10}	0.00051 / 2.6×10^{-10}	0.00037 / 1.9×10^{-10}	0.00037 / 1.9×10^{-10}	0.00010 / 5.0×10^{-11}	0.002 / 1.0×10^{-9}
0.000013 / 6.2×10^{-12}	0.000024 / 1.2×10^{-11}	0.000013 / 6.2×10^{-12}	0.000013 / 6.2×10^{-12}	0.000011 / 5.5×10^{-12}	0.000051 / 2.6×10^{-11}
22 / 0.0088	38 / 0.015	22 / 0.0088	22 / 0.0088	50 / 0.02	22 / 0.0088
60 / 0.000024	250 / 0.0001	60 / 0.000024	60 / 0.000024	500 / 0.0002	60 / 0.000024
Dose / LCF	Dose / LCF	Dose / LCF	Dose / LCF	Dose / LCF	Dose / LCF
0.017 / 8.3×10^{-6}	0.02 / 0.00001	0.017 / 8.3×10^{-6}	0.017 / 8.3×10^{-6}	0.023 / 0.000011	0.024 / 0.000012
0.0021 / 1.0×10^{-9}	0.00051 / 2.6×10^{-10}	0.0021 / 1.0×10^{-9}	0.0021 / 1.0×10^{-9}	0.0003 / 1.5×10^{-10}	0.004 / 2.0×10^{-9}
231 / 0.092	38 / 0.015	319 / 0.13	231 / 0.092	150 / 0.06	297 / 0.12
None	None	None	None	None	None
5.6×10^{-6} (DBA); 0.000013 (BDBA)	0.00014 (DBA)	0.00022 (DBA); 0.000013 (BDBA)	5.6×10^{-6} (DBA); 0.000013 (BDBA)	0.011	0.0090 (DBA)
4.8×10^{-8} (DBA); 2.2×10^{-7} (BDBA)	7.2×10^{-8} (DBA)	1.9×10^{-6} (DBA); 2.2×10^{-7} (BDBA)	4.8×10^{-8} (DBA); 2.2×10^{-7} (BDBA)	6.6×10^{-6}	0.000076 (DBA)
1.5×10^{-8} (DBA); 1.5×10^{-9} (BDBA)	6.2×10^{-7} (DBA)	4.9×10^{-8} (DBA); 1.5×10^{-9} (BDBA)	1.5×10^{-8} (DBA); 1.5×10^{-9} (BDBA)	3.4×10^{-7}	2.7×10^{-6} (DBA)
Less than ERPG-1	Less than ERPG-1	Less than ERPG-1	Less than ERPG-1	Less than ERPG-1	Less than ERPG-1
Less than ERPG-1	Less than ERPG-1	Less than ERPG-1	Less than ERPG-1	Less than ERPG-1	Less than ERPG-1
No disproportionately high and adverse impacts to minority or low-income populations					
23.6 (18 at ANL-W; 5.6 at SRS)		63.6	94.62 (18 at ANL-W; 76.62 at SRS)		86
2,960.5 (770.5 at ANL-W; 2,190 at SRS)		845	1,178.5 (770.5 at ANL-W; 408 at SRS)		924
100.7 (10.7 at ANL-W; 90 at SRS)		12.8	27.2 (10.7 at ANL-W; 16.5 at SRS)		14.1
Person-rem / LCF	Person-rem / LCF	Person-rem / LCF	Person-rem / LCF	Person-rem / LCF	Person-rem / LCF
0.03 / 0.000015	0.0012 / 6.0×10^{-6}	0.14 / 0.000072	0.03 / 0.000015	0.0012 / 6.0×10^{-6}	0.2 / 0.0001
0.004 / 1.6×10^{-6}	0.0012 / 4.7×10^{-7}	0.02 / 7.9×10^{-6}	0.004 / 1.6×10^{-6}	0.0012 / 4.7×10^{-7}	0.027 / 0.000011
less than 1.0×10^{-6} / less than 1.0×10^{-9}	less than 3.0×10^{-6} / less than 1.5×10^{-9}	less than 1.0×10^{-6} / less than 1.0×10^{-9}	less than 1.0×10^{-6} / less than 1.0×10^{-9}	less than 3.0×10^{-6} / less than 1.5×10^{-9}	less than 1.0×10^{-6} / less than 1.0×10^{-9}

[c] Annual dose represents the maximum dose in a single year. Population doses (population and worker population) are in person-rem; individual doses are in millirem. The regulatory dose limit for offsite individuals (public) is 10 millirem per year from air exposures and 100 millirem per year for all pathways. The administrative control limit for an individual worker at a DOE site is 2,000 millirem per year.

[d] Includes 142 cubic meters of spent nuclear fuel.

[e] Includes 25.2 cubic meters of spent nuclear fuel.

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Chapter 3

Affected Environment

3. AFFECTED ENVIRONMENT

Chapter 3 provides an overview of the affected environment of the alternative sites under consideration for the treatment and management of sodium-bonded spent nuclear fuel. The chapter first addresses the approach to defining the affected environment, and then provides a discussion of the affected environment at the Idaho National Engineering and Environmental Laboratory and the Savannah River Site. The discussion of each resource area at each site initially addresses the site as a whole, followed by a description of the proposed treatment locations.

3.1 Approach to Defining the Affected Environment

In accordance with the Council on Environmental Quality guidance under National Environmental Policy Act (NEPA) regulations (40 CFR 1500 through 1508) for preparing an environmental impact statement (EIS), the affected environment is "interpreted comprehensively to include the natural and physical environment and the relationship of people with that environment." The affected environment descriptions presented in this chapter provide the context for understanding the environmental consequences described in Chapter 4. They serve as a baseline for identifying and evaluating the environmental changes that may result from implementing any of the alternatives.

Candidate sites for the treatment and management of sodium-bonded spent nuclear fuel include the U.S. Department of Energy's (DOE) Argonne National Laboratory-West (ANL-W), located within the boundaries of the Idaho National Engineering and Environmental Laboratory (INEEL), and the Savannah River Site's (SRS) F-Area and L-Area. The affected environment is described for the following resource areas: land use, site infrastructure, air quality and noise, water resources, geology and soils, ecological resources, cultural and paleontological resources, socioeconomics, environmental justice, existing human health risk, and waste management. For each DOE site, each resource area is described first for the site as a whole and then for the candidate treatment sites, as appropriate. The level of detail varies depending on the potential for impacts resulting from each treatment and management alternative.

The affected environment for each candidate site presented in this section is based on the *Surplus Plutonium Disposition Final Environmental Impact Statement* (DOE 1999h), unless otherwise noted. Additional information on the affected environment was determined from other recent EISs, previous environmental studies, relevant laws and regulations, and other government reports and databases. More detailed information on the affected environment at the candidate sites can be found in annual site environmental reports and site NEPA documents such as the *Idaho National Engineering and Environmental Laboratory Advanced Mixed Waste Treatment Project Final Environmental Impact Statement* (DOE 1999a) and the *Savannah River Site Spent Nuclear Fuel Management Final Environmental Impact Statement* (DOE 2000).

3.2 Idaho National Engineering and Environmental Laboratory

INEEL is located on approximately 230,700 hectares (570,000 acres) in southeastern Idaho and is 55 kilometers (34 miles) west of Idaho Falls; 61 kilometers (38 miles) northwest of Blackfoot; and 35 kilometers (22 miles) east of Arco. INEEL is owned by the Federal Government and administered, managed, and controlled by DOE. It is primarily within Butte County, but portions of the site are also in Bingham, Jefferson, Bonneville, and Clark counties. The site is roughly equidistant from Salt Lake City, Utah, and Boise, Idaho.

There are 450 buildings and 2,000 support structures at INEEL, with more than 279,000 square meters (3,000,000 square feet) of floor space in varying conditions of utility. INEEL has approximately 25,100 square meters (270,000 square feet) of covered warehouse space and an additional 18,600 square meters (200,000 square feet) of fenced yard space. The total area of the various machine shops is 3,035 square meters (32,665 square feet).

Fifty-two research and test reactors have been used at INEEL over the years to test reactor systems, fuel and target design, and overall safety. In addition to nuclear research reactors, other INEEL facilities are operated to support reactor operations. These facilities include high- and low-level radioactive waste processing and storage sites; hot cells; analytical laboratories; machine shops; and laundry, railroad, and administrative facilities. Other activities include management of one of DOE's largest storage sites for low-level radioactive waste and transuranic waste.

3.2.1 Land Resources

3.2.1.1 Land Use

The Federal Government, the State of Idaho, and private parties own lands surrounding INEEL. Regional land uses include grazing, wildlife management, rangeland, mineral and energy production, recreation, and crop production. Approximately 60 percent of the surrounding area is used by sheep and cattle for grazing. Small communities and towns near the INEEL boundaries include Mud Lake to the east; Arco, Butte City, and Howe to the west; and Atomic City to the south. Two national natural landmarks border INEEL: Big Southern Butte (2.4 kilometers [1.5 miles] south) and Hell's Half Acre (2.6 kilometers [1.6 miles] southeast). A portion of Hell's Half Acre National Natural Landmark is designated as a Wilderness Study Area. The Black Canyon Wilderness Study Area also is adjacent to the northwest boundary of INEEL.

Land-use categories at INEEL include facility operations, grazing, general open space, and infrastructure (e.g., roads). Generalized land uses at INEEL and within the vicinity are shown in **Figure 3–1.** Facility operations include industrial and support operations associated with energy research and waste management activities. Up to 340,000 acres (137,600 hectares) of the site is leased for cattle and sheep grazing; grazing permits are administered by the Bureau of Land Management (DOE 1999i). Land also is used for recreation and environmental research associated with the designation of INEEL as a National Environmental Research Park. Much of INEEL is open space that has not been designated for specific use. Some of this space serves as a buffer zone between INEEL facilities and other land uses. Recently, approximately 29,950 hectares (74,000 acres) of open space in the north central portion of the site have been designated as the INEEL Sagebrush Steppe Ecosystem Reserve (DOE 1999g). This area represents some of the last sagebrush steppe ecosystem in the United States and provides a home for a number of rare and sensitive species of plants and animals. About 2 percent of the total INEEL site area (4,600 hectares [11,400 acres]) is used for facilities and operation. INEEL facilities are sited within a central core area of about 93,100 hectares (230,000 acres) (Figure 3-1). Public access to most facilities is restricted. Approximately 6 percent of INEEL (34,000 acres [13,760 hectares]) is devoted to utility rights-of-way and public roads (DOE 1999i). DOE land-use plans and policies applicable to INEEL are discussed 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 Spent Nuclear Fuel EIS) (DOE 1995a).

The total land area at ANL-W is 328 hectares (810 acres); however, site facilities cover only about 20 hectares (50 acres), or 6 percent of the site (DOE 1996a). ANL-W is located 7 kilometers (4.3 miles) northwest of the nearest site boundary and is designated as a testing center for advanced technologies associated with nuclear power systems. The area has 52 major buildings, including reactor buildings, laboratories, warehouses, technical and administrative support buildings, and craft shops that comprise 55,700 square meters (600,000 square feet) of floor space (LMITCO 1997). Five nuclear test reactors, including the Experimental

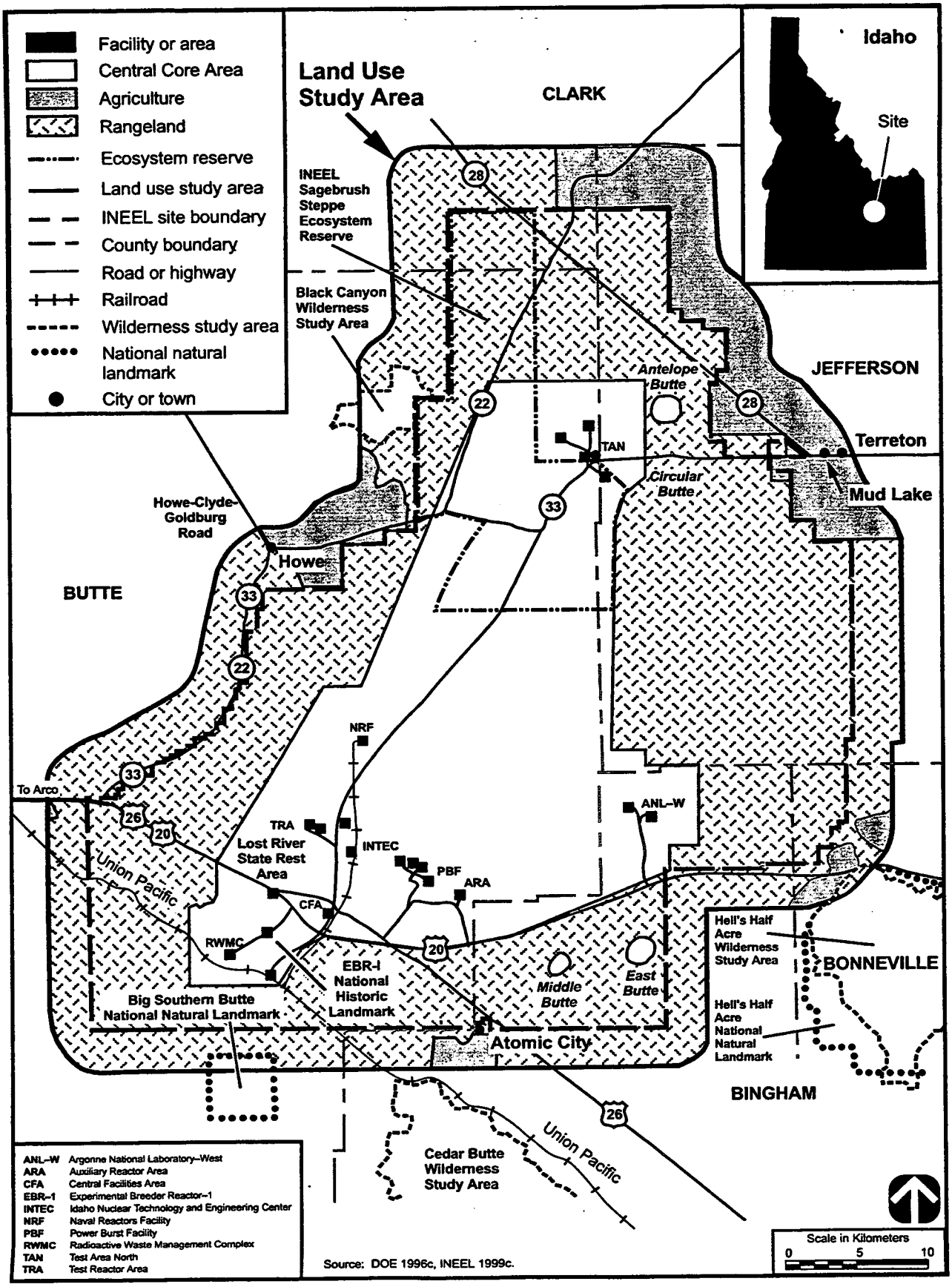

Figure 3–1 Generalized Land Use at INEEL and Vicinity

Breeder Reactor II (EBR-II), have operated on the site, although the only one currently active is a small reactor used for radiography examination of experiments, waste containers, and spent nuclear fuel. The Fuel Conditioning Facility and the Hot Fuel Examination facility also are located at the site (DOE 1996a).

3.2.1.2 Visual Resources

The Bitterroot, Lemhi, and Lost River mountain ranges border the INEEL site on the north and west. Volcanic buttes near the southern boundary of INEEL can be seen from most locations on the site. Lands adjacent to the site under Bureau of Land Management jurisdiction are designated as Visual Resource Management Class II areas. However, the Black Canyon Wilderness Study Area, located adjacent to the northwest site boundary, is under consideration by the Bureau of Land Management for Wilderness designation. If approved, this area would upgrade its Visual Resource Management rating to Class I. INEEL itself generally consists of open desert land mostly covered by big sagebrush and grasslands. Most of the land within the site falls within Visual Resource Management Class II and III. Management activities within these classes may be seen but should not dominate the view (DOI 1986).

Ten facility areas are located on the INEEL site. Although INEEL has a comprehensive facility and land use plan (LMITCO 1997), no specific visual resource standards have been established. INEEL facilities appear as low-density commercial/industrial complexes widely dispersed throughout the site. Structure heights range from about 3 to 30 meters (10 to 100 feet); a few stacks and towers reach 76 meters (250 feet). Although many INEEL facilities are visible from highways, most facilities are more than 0.8 kilometers (0.5 miles) from public roads. The operational areas are well defined at night by security lights.

Developed areas within ANL-W are consistent with a Visual Resource Management Class IV rating in which management activities dominate the view and are the focus of viewer attention. The tallest structure at ANL-W is the Fuel Conditioning Facility stack, which is 61 meters (200 feet) in height. The site is visible from Highway 20. Facilities that stand out from the highway include the Hot Fuel Examination Facility, the EBR-II containment shell, the Zero Power Physics Reactor, and the Transient Reactor Test Facility. Natural features of visual interest within a 40-kilometer (25-mile) radius of ANL-W include the East Butte at 9 kilometers (5.6 miles), Middle Butte at 11 kilometers (6.8 miles), Hell's Half Acre National Natural Landmark and Hell's Half Acre Wilderness Study Area at 15 kilometers (9.3 miles), Big Lost River at 19 kilometers (11.8 miles), and Big Southern Butte National Natural Landmark at 30 kilometers (18.6 miles).

3.2.2 Site Infrastructure

Site infrastructure includes those utilities and other resources required to support modification and continued operation of mission-related facilities identified under the various alternative actions. INEEL has extensive production, service, and research facilities. An extensive infrastructure system supports these facilities, as shown in **Table 3–1**.

3.2.2.1 Transportation

The road network at INEEL provides for onsite transportation; railroads are used for deliveries of large volumes of coal and oversized structural components. Commercial shipments are transported by truck; some bulk materials are transported by train; and waste is transported by truck and train. About 140 kilometers (87 miles) of paved surface have been developed out of the 445 kilometers (277 miles) of roads on the site, including 29 kilometers (18 miles) of service roads that are closed to the public. Most of the roads are adequate for the current level of normal transportation activity and could handle increased traffic volume.

Idaho Falls receives railroad freight service from Butte, Montana, to the north, and from Pocatello, Idaho, and Salt Lake City, Utah, to the south. The Union Pacific Railroad's Blackfoot-to-Arco Branch crosses the

southern portion of INEEL and provides rail service to the site. This branch connects with a DOE spur line at Scoville Siding, then links with developed areas within INEEL. There are 48 kilometers (30 miles) of railroad track at INEEL. Rail shipments to and from INEEL usually are limited to bulk commodities, spent nuclear fuel, and radioactive waste.

Table 3–1 INEEL Site-Wide Infrastructure Characteristics

Resource	*Current Usage*	*Site Capacity*
Transportation		
Roads (kilometers)	445 [a]	Not applicable
Railroads (kilometers)	48	Not applicable
Electricity		
Energy consumption (megawatt hours per year)	221,772 [b]	394,200
Peak load (megawatts)	39 [b]	124
Fuel		
Natural gas (cubic meters per year)	Not applicable	Not applicable
Oil and propane (liters per year)	5,820,000	16,000,000 [c]
Coal (metric tons per year)	11,340	11,340 [c]
Water (liters per year)	4,900,000,000 [d]	43,000,000,000 [e]

[a] Includes paved and unpaved roads.
[b] Fiscal Year 1997 data based on INEEL 1998.
[c] Low supplies can be replenished by truck or rail.
[d] 1997 usage based on DOE 1999a.
[e] Water right allocation.
Source: DOE 1999h, except as noted in footnotes b and d.

3.2.2.2 Electricity

Commercial electric power is supplied to INEEL through two feeders from the Antelope substation to the federally owned Scoville substation, which supplies electric power directly to the site's electric power distribution system. Electric power supplied by Idaho Power Company is generated by hydroelectric generators along the Snake River in southern Idaho and by the Bridger and Valmy coal-fired thermal electric generation plants in southwestern Wyoming and northern Nevada.

The average electrical availability at INEEL is about 394,200 megawatt hours per year; in 1997 the average usage was 221,772 megawatt hours. The peak load capacity for INEEL is 124 megawatts; the 1997 peak load usage was about 39 megawatts (INEEL 1998). Current electrical usage at ANL-W is 28,700 megawatt hours per year (Goff 1999).

3.2.2.3 Fuel

Fuel consumed at INEEL include several types of liquid petroleum fuel, coal, and propane gas. All fuel is transported to the site for use and storage. Fuel storage is provided for each facility, and the inventories are restocked as necessary. The current site usage of fuel oil is about 5.8 million liters (1.5 million gallons) per year. The current site usage of coal is about 11,340 metric tons (12,500 tons) per year. If additional coal or fuel oil were needed during the year, it could be shipped to the site.

3.2.2.4 Water

The Snake River Plain Aquifer is the source of all water at INEEL. The water is provided by a system of about 30 wells, together with pumps and storage tanks. That system is administered by DOE, which holds the

Federal Reserved Water Right of 43 billion liters (11.4 billion gallons) per year for the site. The current site usage is about 4.9 billion liters (1.3 billion gallons) per year (DOE 1999a).

3.2.2.5 Site Safety Services

DOE operates three fire stations at INEEL. These stations are at the north end of Test Area North, at ANL-W, and in the Central Facilities Area. Each station has a minimum of one engine company capable of supporting any fire emergency in its assigned area. The fire department also provides the site with ambulance, emergency medical technician, and hazardous material response services.

3.2.3 Air Quality and Noise

3.2.3.1 Air Quality

The climate at INEEL and the surrounding region is characterized as a semiarid steppe with low relative humidity, wide daily temperature swings, and large variations in annual precipitation. The average annual temperature at INEEL is 5.6 °C (42 °F), and average seasonal temperatures range from a minimum of -7.3 °C (18.8 °F) in winter to 18.2 °C (64.8 °F) in summer. Temperature extremes range from a summertime maximum of 39.4 °C (103 °F) to a wintertime minimum of -45 °C (-49 °F). The average annual precipitation at INEEL is 22 centimeters (8.7 inches). Prevailing winds at INEEL are predominantly southwest or northeast, although terrain features may cause variations in the flow (DOE 1999a). The average annual wind speed is 3.4 meters per second (7.5 miles per hour).

Nonradiological Releases

INEEL is within Eastern Idaho Intrastate Air Quality Control Region No. 61. None of the areas within INEEL or its surrounding counties are designated as nonattainment areas, i.e., areas where criteria air pollutant levels exceed the National Ambient Air Quality Standards (NAAQS) established by the U.S. Environmental Protection Agency (EPA) (40 CFR 50). The nearest nonattainment area for particulate matter is in Pocatello, about 80 kilometers (50 miles) to the south. Applicable NAAQS and Idaho State ambient air quality standards are presented in **Table 3–2.**

The primary sources of nonradiological air pollutants at INEEL currently include calcination of sodium-bearing waste, combustion of coal for steam, and combustion of fuel oil for heating. Other emission sources include waste burning, industrial processes, stationary diesel engines, vehicles, and fugitive dust from activities including waste burial and construction. The existing ambient air concentrations attributable to sources at INEEL are presented in Table 3–2. For criteria pollutants, concentrations are based on (1) dispersion modeling at the INEEL site boundary centered at the Idaho Nuclear Technology and Engineering Center (INTEC) facility, performed for the High-Level Waste and Facilities Disposition EIS (DOE 1999i) using 1997 actual emissions and excluding ANL-W; and (2) dispersion modeling at the INEEL site boundary centered on ANL-W using 1997 actual emissions. The modeling performed for the High-Level Waste and Facilities Disposition EIS used EPA's ISCST3 model with hourly meteorological data. The ANL-W modeling used EPA's SCREEN3 model, which is very conservative compared to ISCST3, and uses a set of worst-case meteorological conditions to predict a maximum one-hour concentration. This one-hour concentration was converted to other averaging times using regulatory scaling factors (SCDHEC 1993). For these reasons, the ANL-W concentrations are extremely conservative. In spite of this conservatism, total maximum site boundary concentrations (which can be approximated by summing individual concentrations) are well below ambient air quality standards.

For acrolein, cadmium, toluene, and xylene, the concentrations in Table 3–2 are based on dispersion modeling using actual INEEL site-wide emissions for the year 1997 and modeling these emissions as if they all

originated from ANL-W. For the remaining hazardous/toxic air pollutants, concentrations are based on site-wide dispersion modeling using maximum emissions for the year 1990, as presented in the Programmatic Spent Nuclear Fuel EIS (DOE 1995a). Only those hazardous/toxic air pollutants that would be emitted for any of the alternatives evaluated in this EIS are presented. The hazardous/toxic air pollutant standards are presented for informational purposes only, as the standards apply only to new or modified emissions sources.

Table 3–2 Comparison of Modeled Ambient Air Concentrations at the INEEL Site Boundary From INEEL Sources With Most Stringent Applicable Standards or Guidelines

Pollutant	*Averaging Period*	*Most Stringent Standard or Guideline (micrograms per cubic meter)*	*Maximum INEEL Concentration Excluding ANL-W for Criteria Pollutants* [a] *(micrograms per cubic meter)*	*Maximum ANL-W Concentration* [b] *(micrograms per cubic meter)*
Criteria Pollutants [a]				
Carbon monoxide	8 hours 1 hour	10,000 [c] 40,000 [c]	78 206	41 59
Nitrogen dioxide	Annual	100 [c]	0.46	13
Ozone	8 hours	157 [d]	(e)	(e)
PM_{10}	Annual 24 hours (interim) 24 hours (99th percentile over 3 years)	50 [c] 150 [c] 150 [d]	0.49 12 (f)	0.14 1.1 (f)
$PM_{2.5}$	3-year annual 24 hours (98th percentile over 3 years)	15 [d] 65 [d]	(f) (f)	(f) (f)
Sulfur dioxide	Annual 24 hours 3 hours	80 [c] 365 [c] 1,300 [c]	0.14 5.3 24	3.3 27 60
Hazardous/Toxic Air Pollutants [g]				
1,3-Butadiene	Annual	0.0036	0.001	Not applicable
Acetaldehyde	Annual	0.45	0.0110	Not applicable
Acrolein [h]	24 hours	12.5	0.00332	Not applicable
Benzene	Annual	0.12	0.0290	Not applicable
Cadmium [h]	Annual	0.00056	0.0000415	Not applicable
Formaldehyde	Annual	0.077	0.012	Not applicable
Toluene [h]	24 hours	18,750	0.392	Not applicable
Xylene [h]	24 hours	21,750	0.0362	Not applicable

PM_n = Particulate matter less than or equal to n microns in diameter.

[a] Concentrations for criteria pollutants from the High-Level Waste and Facilities Disposition EIS, based on actual emissions plus projected increases, for dispersion modeling centered on INTEC with no contribution from ANL-W (DOE 1999i: Table C 2-14, No Action Alternative). Concentrations for hazardous and toxic compounds were estimated based on site-wide modeling of 1990 emissions, as presented in the Programmatic Spent Nuclear Fuel EIS.

[b] Concentrations for criteria pollutants based on dispersion modeling centered on ANL-W, using 1997 ANL-W actual emissions.

[c] Federal and state standards.

[d] Standard currently under litigation, but will become enforceable during the life of the project.

[e] Not directly emitted or monitored by the site.

[f] No data are available with which to assess particulate matter concentrations.

[g] Note that standards apply only to new or modified sources and are provided for informational purposes only.

[h] Estimated based on 1997 INEEL emissions, modeling all emissions as if they originated from ANL-W.

Sources: 40 CFR 50, Rules 577, 585, 586; ID APA 16.01.01, DOE 1995a, DOE 1999i, 62 FR 38855, 62 FR 38652.

The nearest Prevention of Significant Deterioration Class I area[1] to INEEL is Craters of the Moon Wilderness Area, Idaho, located 53 kilometers (33 miles) west-southwest from the center of the site. There are no other Class I areas within 100 kilometers (62 miles) of INEEL. INEEL and its vicinity are classified as a Prevention of Significant Deterioration Class II area[2].

The EPA has established Prevention of Significant Deterioration increments for certain pollutants: sulfur dioxide, nitrogen dioxide, and particulate matter less than or equal to 10 microns in diameter (PM_{10}). The increments specify a maximum allowable increase above a certain baseline concentration for a given averaging period, and apply only to sources constructed or modified after a specified baseline date. These sources are known as increment-consuming sources. The baseline date is the date of submittal of the first application for a Prevention of Significant Deterioration permit in a given area.

Prevention of Significant Deterioration permits have been obtained for the coal-fired, steam-generating facility (located next to INTEC) and the Fuel Processing Facility. Operation of the Fuel Processing Facility is not expected (DOE 1996c). In addition to these facilities, INEEL has other increment-consuming sources on site. **Tables 3–3** and **3–4** specify the current amount of Prevention of Significant Deterioration increment consumption in Class I and Class II areas, respectively, by INEEL's increment-consuming sources based on dispersion modeling analyses.

Table 3–3 Prevention of Significant Deterioration Increment Consumption at Craters of the Moon Wilderness (Class I) Area by Existing (1996) and Projected Sources Subject to Prevention of Significant Deterioration Regulation[a]

Pollutant	*Averaging Time*	*Allowable Prevention of Significant Deterioration Increment[b] (micrograms per cubic meter)*	*Amount of Prevention of Significant Deterioration Increment Consumed (micrograms per cubic meter)*
Nitrogen dioxide[c]	Annual	2.5	0.004
Respirable particulates[d]	Annual	4	0.008
	24 hours	8	0.6
Sulfur dioxide	Annual	2	0.09
	24 hours	5	1.8
	3 hours	25	5.9

[a] Projected sources include emissions between the present and the time that the proposed Advanced Mixed Waste Treatment Facility becomes operational.

[b] All increments specified are State of Idaho standards (ID DHW Rule 581; ID APA 16.01.01).

[c] Assumes that the New Waste Calcining Facility (the largest source of nitrogen dioxide emissions at INEEL) operates for the entire year.

[d] Data on particulate size are not available for most sources. For purposes of comparison to the respirable particulate increments, it is assumed conservatively that all particulates emitted are of respirable size (i.e., 10 microns or less in diameter).

Source: DOE 1999a.

Routine offsite monitoring for nonradiological air pollutants generally is performed only for particulates. Monitoring for PM_{10} is performed by the Environmental Science and Research Foundation at the site boundary and at communities beyond the boundary. In 1997, 49 samples were collected at Rexburg (located about 65 kilometers [40 miles] east of the site). The mean PM_{10} concentration at Rexburg was 14 micrograms per cubic meter. Forty-one samples were collected at the Mountain View Middle School in Blackfoot in 1997, with a mean concentration of 15 micrograms per cubic meter. Twenty-nine samples were collected at Atomic City in 1997, with a mean concentration of 15 micrograms per cubic meter (Evans et al. 1998).

[1] *Class I areas are defined as national parks and wildlife refuges.*

[2] *Class II areas are defined as any area not designated Class I. Please see Appendix B, Impact Assessment Methods, for a more detailed discussion.*

Table 3–4 Prevention of Significant Deterioration Increment Consumption at Class II Areas by Existing (1996) and Projected Sources Subject to Prevention of Significant Deterioration Regulation at INEEL [a]

Pollutant	*Averaging Time*	*Allowable Prevention of Significant Deterioration Increment [b] (micrograms per cubic meter)*	*Amount of Prevention of Significant Deterioration Increment Consumed [c] (micrograms per cubic meter)*
Nitrogen dioxide [d]	Annual	25	1.3
Respirable particulates [e]	Annual	17	0.1
	24 hours	30	3.8
Sulfur dioxide	Annual	20	1.8
	24 hours	91	12
	3 hours	512	74

[a] Projected sources include emissions between the present and the time that the proposed Advanced Mixed Waste Treatment Facility becomes operational.
[b] All increments specified are State of Idaho standards (ID DHW Rule 581; ID APA 16.01.01).
[c] Maximum concentration predicted at the INEEL site boundary.
[d] Assumes that the New Waste Calcining Facility operates for the entire year.
[e] Data on particulate size are not available for most sources. For purposes of comparison to the respirable particulate increments, it is assumed conservatively that all particulates emitted are of respirable size (i.e., 10 microns or less in diameter).
Source: DOE 1999a.

Some monitoring data also have been collected by the National Park Service at the Craters of the Moon Wilderness Area. The monitoring program has shown no cases in which the primary 1-hour ozone and total suspended particulate standards and low levels of sulfur dioxide were exceeded (except for one case in which the 24-hour standard was exceeded in 1985) (DOE 1999a). Note that the total suspended particulates within standards have been replaced with PM_{10} standards and the 1-hour ozone standard has been replaced by the 8-hour standard.

The primary sources of nonradiological air emissions at ANL-W include four water tube boilers for site heating and process requirements, various emergency or standby diesel generators used for backup power, a permitted paint spray booth, a permitted decontamination facility at the Fuel Conditioning Facility, and two fixed-roof storage tanks that hold fuel for the boilers (DOE 1998a).

Radiological Releases

The primary sources of radiological air pollutants at INEEL as well as the localized releases at ANL-W are presented in **Table 3–5**. During 1997, an estimated 5,596 curies of radioactivity were released to the atmosphere from all INEEL sources. Ninety-five percent of the total airborne radioactive effluent was released from two INEEL facilities, the Test Reactor Area and ANL-W. The Test Reactor Area released a total of 1,695 curies, of which over 93 percent was from radioisotopes of noble gases. ANL-W released 3,605 curies, of which radioisotopes of noble gases comprised over 99 percent.

Year-to-year fluctuations in airborne radioactive effluent releases are dependent on which processes are active at INEEL facilities. The total for 1997 is considerably higher than the annual totals for 1993 to 1996, due primarily to the 3,579 curies of krypton-85 released from ANL-W. Krypton-85, a noble gas, was released from ANL-W as part of a spent fuel treatment project, the Electrometallurgical Treatment Research and Demonstration Project in the Fuel Conditioning Facility at Argonne National Laboratory-West. Although these 1997 releases are higher than previous years, they are still considerably less than the annual totals for the 1980s.

Table 3–5 Radiological Gaseous and Airborne Emissions at INEEL in 1997 (Curies)

Emission Type	*Radionuclide* [a]	*ANL-W*	*Other Facilities at INEEL* [b]	*Total* [c]
Noble gases	Argon-41	3.9	1,550	1,554
	Krypton-85	3,579	—	3,579
	Xenon-135	—	20.9	20.9
	Krypton-88	—	3.5	3.5
	Krypton-85m	—	3.0	3.0
	Xenon-133	—	3.4	3.4
	Krypton-87	—	1.8	1.8
	Xenon-138	—	0.8	0.8
	Xenon-135m	—	1.1	1.1
Airborne particulates	Rubidium-88	—	1.3	1.3
	Rubidium-89	—	0.011	0.011
	Cesium-138	—	0.069	0.069
	Chromium-51	—	0.0056	0.0056
	Sodium-24	—	0.014	0.0056
	Cesium-137	—	0.0071	0.0071
	Technetium-99m	—	0.0022	0.0022
	Antimony-125	—	0.000027	0.000027
	Strontium-90 [d]	—	0.00070	0.00070
	Plutonium-238	—	5.1×10^{-6}	5.1×10^{-6}
	Plutonium-239	1.1×10^{-7}	1.5×10^{-6}	1.6×10^{-6}
Tritium, C-14, and iodine isotopes	Tritium (H-3)	23.0	403	426
	Carbon-14	—	0.91	0.91
	Iodine-129	—	0.058	0.058
	Iodine-131	—	0.0017	0.0017
	Iodine-133	—	0.00055	0.00055
Others		0.000039	0.0035	0.0035
Totals		**3,606**	**1,990**	**5,596**

[a] The table includes all radionuclides with total releases greater than 10^{-7} curies. Values are not corrected for decay after release.
[b] Facilities include INTEC, the Test Reactor Area, Naval Reactor Facility, Central Facility Area, Radioactive Waste Management Complex, and Power Burst Facility.
[c] Rounded totals include small amounts from facilities not listed.
[d] Parent-daughter equilibrium assumed.
Source: Evans et al. 1998.

3.2.3.2 Noise

Major noise emission sources within INEEL include various industrial facilities, equipment, and machines. Most INEEL industrial facilities are far enough from the site boundary that noise levels at the boundary would not be measurable or would be barely distinguishable from background levels.

Existing INEEL-related noises of public significance are from the transportation of people and materials to and from the site and in-town facilities via buses, trucks, private vehicles, helicopters, and freight trains. Noise measurements recorded 15 meters (50 feet) from U.S. Route 20 indicate that the sound levels from traffic range from 64 to 86 decibels A-weighted, and that the primary source is buses (71 to 80 decibels A-weighted).

While few people reside within 15 meters (50 feet) of the roadway, the results indicate that INEEL traffic noise might be objectionable to members of the public residing near principal highways or busy bus routes. Noise levels along these routes may have decreased somewhat because of reductions in employment and bus service at INEEL in the last few years. The acoustic environment along the INEEL site boundary in rural areas and at nearby areas away from traffic noise is typical of a rural location: the day-night average sound level is in the range of 35 to 50 decibels A-weighted (DOE 1999h). The noise generated at INEEL is not propagated at detectable levels off site, since all public areas are at least 2.5 kilometers (4 miles) away from site facilities.

No distinguishing noise characteristics at ANL-W have been identified. ANL-W is 9 kilometers (5.6 miles) from the site boundary; thus, the contributions from the area to noise levels at the site boundary are not measurable.

3.2.4 Water Resources

3.2.4.1 Surface Water

Three intermittent streams drain the mountains near INEEL: Big Lost River, Little Lost River, and Birch Creek (**Figure 3–2**). These intermittent streams carry snowmelt in the spring and are usually dry by midsummer. Several years can pass before any offsite waters enter DOE property. Big Lost River and Birch Creek are the only streams that regularly flow onto the INEEL site. Little Lost River is usually dry by the time it reaches the site because of upstream use of the flow for irrigation. None of the streams flow from the site to offsite areas. The Big Lost River discharges into the Big Lost River sinks, and there is no surface discharge from these sinks (Barghusen and Feit 1995, DOE 1996c).

The Big Lost River has been classified by the State of Idaho for domestic and agricultural use, cold water biota development, salmon spawning, primary and secondary recreation, and other special resource uses. Surface waters, however, are not used for drinking water on the site, nor is effluent discharged directly to them. Since INEEL facilities currently do not discharge directly to nor make withdrawals from these water bodies, there are no surface water rights issues at INEEL. None of the rivers have been classified as a Wild and Scenic River (DOE 1995a, DOE 1996c).

A preliminary study of the 100-year peak flow for the Big Lost River has been completed by the U.S. Geological Survey (USGS 1998). Additional studies of the 100-year and 500-year flood plains were conducted by the Bureau of Reclamation for DOE (DOE 1999i).

There are no named streams within the ANL-W area and no permanent, natural, surface water features near the area (ANL 1998a). Neither the 100-year flood nor flooding scenarios that involve the failure of Mackay Dam on the Big Lost River indicate that flood waters would reach ANL-W (Koslow and Van Haaften 1986, USGS 1998, DOE 1999i) (**Figure 3–3**).

Nonradiological Releases

ANL-W discharges 11,900,000 liters (3,140,000 gallons) per year of nonhazardous liquid waste to the sewage pond and 68,000,000 liters (18,000,000 gallons) per year to the industrial waste pond (ANL 1999b). These are evaporation ponds and water levels may be controlled by land spreading if necessary (Cascade Earth Sciences 1998).

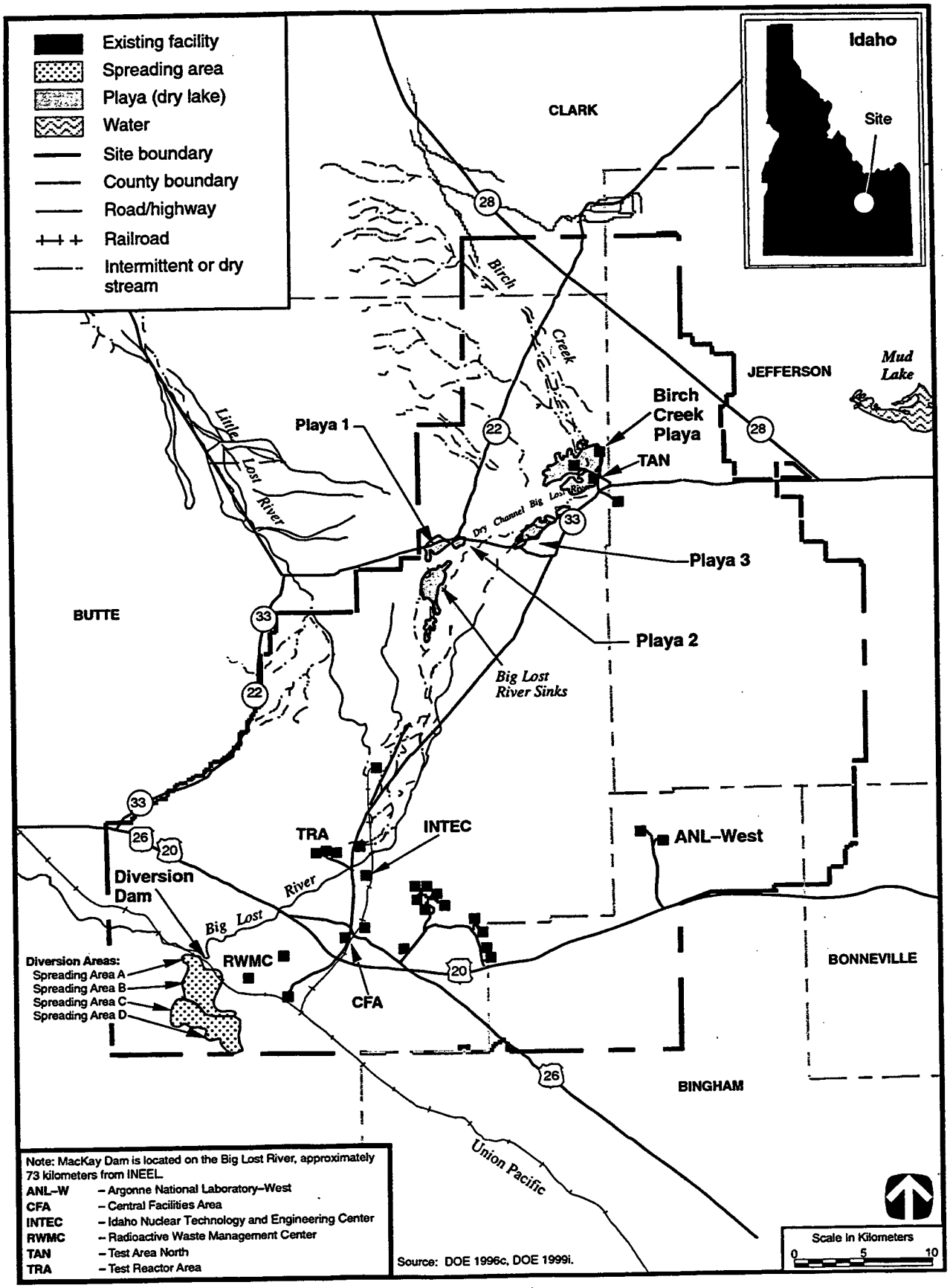

Figure 3–2 Surface Water Features at INEEL

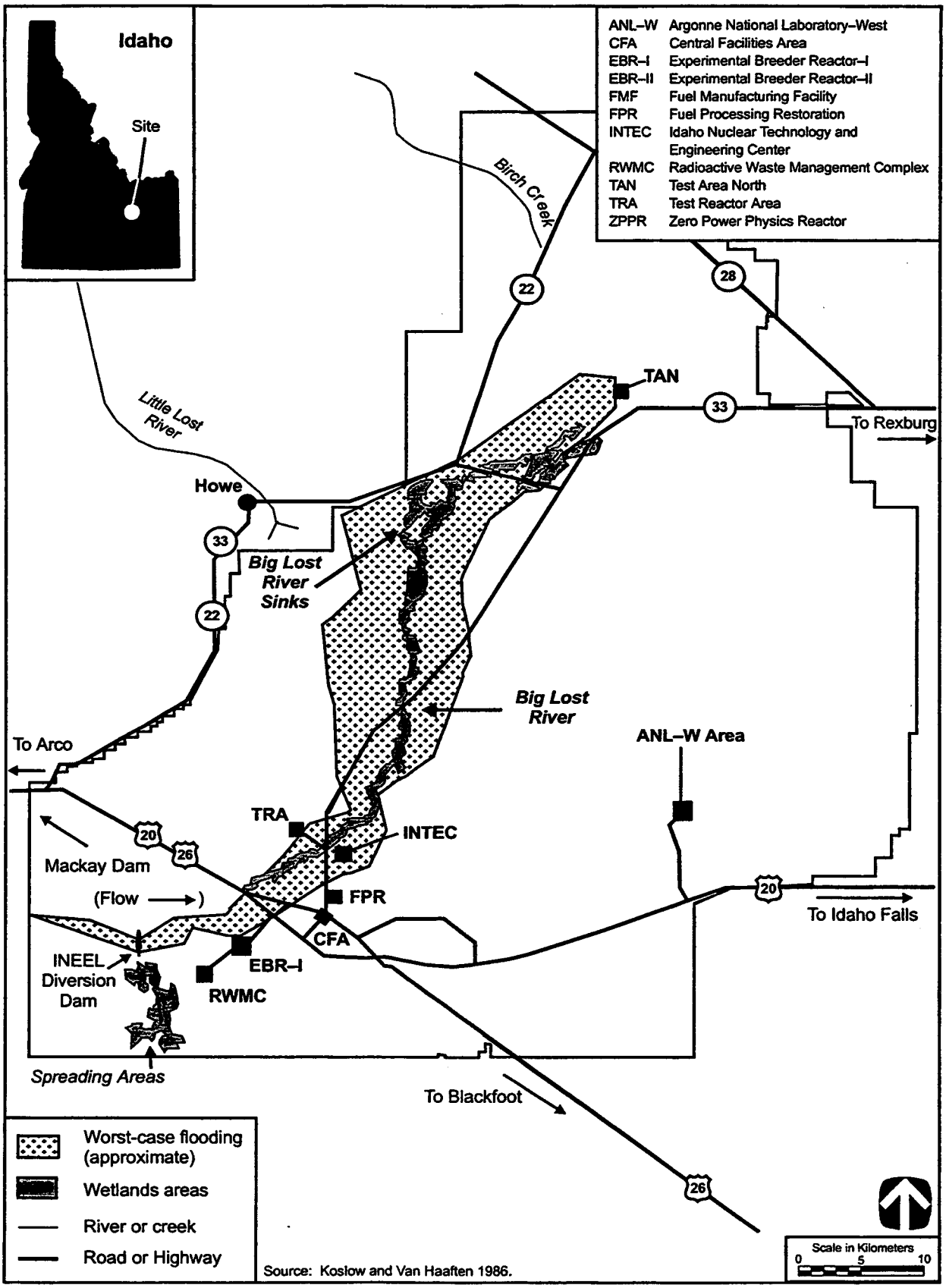

Figure 3–3 Flood Area for the Probable Maximum Flood-Induced Overtopping Failure of the Mackay Dam

Radiological Releases

Table 3–6 summarizes the radioactive liquid effluent released on site during 1997. Virtually all of the 1997 radioactive liquid effluent was released from the Test Reactor Area into two hypalon plastic-lined evaporation ponds which have been in use since August 1993. These ponds serve to prevent contaminant percolation into the ground, thus minimizing contaminant dispersal. No radioactive liquid effluent was released to the offsite environment from INEEL facilities during 1997. Routine injections of radioactive liquid effluent into the Snake River Plain Aquifer ceased in 1984 (Evans et al. 1998).

Table 3–6 Radiological Liquid Effluent Released at INEEL in 1997 (Curies)

Radionuclide [a]	*ANL-W*	*Other Facilities at INEEL* [b]	*Total*
Tritium (H-3)	—	96.3	96.3
Chromium-51	—	2.4	2.4
Cobalt-60	—	0.4	0.4
Hafnium-181	—	0.081	0.081
Strontium-90	—	0.031	0.031
Cesium-137	—	0.017	0.017
Plutonium-239	—	0.0035	0.0035
Gross Beta [c]	—	0.50	0.50
All others	—	0.06	0.06
Total	—	**99.8**	**99.8**

[a] The table includes all radionuclides with total releases greater than 0.001 curies. Values are not corrected for decay after release.
[b] Facilities include INTEC and the Test Reactor Area.
[c] Gross beta assumed to be radioactive strontium.
Source: Evans et al. 1998.

3.2.4.2 Groundwater

Aquifers are classified by Federal and state authorities according to use and quality. The Federal classifications include Class I, II, and III groundwater. Class I groundwater is either the sole source of drinking water or is ecologically vital. Class IIA and IIB are current or potential sources of drinking water (or other beneficial use), respectively. Class III is not considered a potential source of drinking water and is of limited beneficial use.

The Snake River Plain aquifer is classified by the EPA as a Class I sole source aquifer. It lies below the INEEL site and covers about 2,486,000 hectares (6,143,000 acres) in southeastern Idaho. This aquifer serves as the primary drinking water source in the Snake River Basin and is believed to contain 1.2 quadrillion to 2.5 quadrillion liters (317 trillion to 660 trillion gallons) of water. Recharge of the groundwater comes from Henry's Fork of the Snake River, Big Lost River, Little Lost River, and Birch Creek. Rainfall and snowmelt also contribute to the aquifer's recharge.

Groundwater generally flows laterally at a rate of 1.5 to 6.1 meters (5 to 20 feet) per day. It emerges in springs along the Snake River from Milner to Bliss, Idaho (DOE 1996c). Depth to the groundwater table ranges from about 61 meters (200 feet) below ground in the northeast corner of INEEL to about 275 meters (900 feet) in the southeast corner (DOE 1999i). Perched water tables (i.e., bodies of groundwater lying above a more extensive aquifer) occur below the surface. These perched water tables tend to slow the migration of pollutants that might otherwise reach the Snake River Plain aquifer.

INEEL has a large network of monitoring wells—about 120 in the Snake River Plain aquifer and another 100 drilled in the perched zone. The wells are used for monitoring to determine the compliance of specific actions with requirements of the Resource Conservation and Recovery Act (RCRA) and the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA), as well as routine monitoring to evaluate the quality of the water in the aquifer. The Snake River Plain aquifer is known to have been contaminated with tritium; however, the concentration dropped 93 percent between 1961 and 1994, possibly because of the elimination of tritium disposal, radioactive decay, and dispersion throughout the aquifer. Other known contaminants include cesium-137, iodine-129, strontium-90, and nonradioactive compounds such as trichloroethylene, chromium, and sodium. Components of nonradioactive waste have entered the aquifer as a result of past waste disposal practices. Elimination of groundwater injection, except for stormwater management and heat exchange, illustrates a change in disposal practices that has reduced the amount of these constituents in the groundwater. Information on recent groundwater monitoring and chemical analysis is presented in the annual site environmental report (Evans et al. 1998).

In 1997, INEEL used about 4.9 billion liters (1.3 billion gallons) per year from the Snake River Plain aquifer, the only source of water at INEEL (DOE 1999a). This represents less than 0.3 percent of the groundwater withdrawn from that aquifer. DOE holds a Federal Reserved Water Right for the INEEL site that permits a pumping capacity of 2.3 cubic meters (80 cubic feet) per second with a maximum water consumption of 43 billion liters (11.4 billion gallons) per year. INEEL's priority on water rights dates back to its establishment in 1950 (DOE 1996c).

All water used at ANL-W is groundwater from the Snake River Plain aquifer. The depth of the groundwater at ANL-W is approximately 195 meters (640 feet) and the flow is generally to the south-southwest. ANL-W uses approximately 188 million liters (49.6 million gallons) per year of water (ANL 1999b, Cascade Earth Sciences 1998).

No significant levels of radioactivity are found in the production wells at ANL-W. Constituents measured in the groundwater monitoring wells in 1997 were all below regulatory levels (ANL 1998b).

3.2.5 Geology and Soils

INEEL is located on the northwestern edge of the Eastern Snake River Plain that is bounded on the north and south by north to northwest-tending mountains and valley of the Basin and Range Provence (DOE 1999a). The upper 1 to 2 kilometers (0.6 to 1.2 miles) of the crust beneath INEEL is composed of interlayered basalt and sediment. The sediments are composed of fine-grained silts that were deposited by wind; silts, sands, and gravels deposited by streams; and clays, silts, and sands deposited in lakes. Rhyolitic (granite-like) volcanic rocks of unknown thickness lie beneath the basalt sediment sequence. The rhyolitic volcanic rocks erupted between 6.5 and 4.3 million years ago (Barghusen and Feit 1995). Lava tubes, which could have similar adverse effects as karst, occur in the INEEL area (Abbott, Crockett, and Moor 1997).

Within INEEL, economically viable sand, gravel, and pumice resources have been identified. Several quarries have supplied these materials to various onsite construction projects. Geothermal resources are potentially available in parts of the Eastern Snake River Plain, but neither of two boreholes drilled near INTEC encountered rocks with significant geothermal potential.

The Arco Segment of the Lost River Fault terminates about 12 kilometers (7.5 miles) from the INEEL boundary. The South Creek Segment of the Lemhi Fault terminates at the northwest boundary of the site. Both segments are considered capable (Abbott, Crockett, and Moor 1997). A capable fault is one that has had movement at or near the ground surface at least once within the past 35,000 years, or recurrent movement within the past 500,000 years.

The only recent earthquake activity recorded on or in the immediate vicinity of INEEL has been confined to several small- magnitude microearthquakes (less than 1.5 on the Richter Scale) (Jackson et al. 1993). The largest historic earthquake near INEEL took place in 1983, 107 kilometers (66 miles) to the northwest, near Borah Peak in the Lost River Range. The earthquake had a moment magnitude of 6.9 with a ground acceleration of 0.022 g to 0.078 g at INEEL (Jackson 1985). An earthquake with a maximum horizontal acceleration of 0.15 g is calculated to have an annual probability of occurrence of 1 in 5,000 at a central INEEL location (Barghusen and Feit 1995).

Basaltic volcanic activity occurred near INEEL from about 4 million to 2,100 years ago. Although no eruptions have occurred on the Eastern Snake River Plain during recorded history, lava flows of the Hell's Half Acre lava field erupted near the southern INEEL boundary as recently as 5,400 years ago. The most recent eruptions within the site area occurred about 2,100 years ago 30 kilometers (19 miles) southwest of the site at the Craters of the Moon Wilderness Area. Five volcanic zones have been identified on INEEL. The estimated recurrence interval for volcanism in these zones ranges from 16,000 to 100,000 years (Hackett and Smith 1994).

Four basic soilscapes exist on INEEL: river-transported sediments deposited on alluvial plains; fine-grained sediments eroded into lake or playa basins; colluvial sediments originating from bordering mountains; and wind-blown sediments over lava flows. The alluvial deposits follow the courses of the modern Big Lost River and Birch Creek. The playa soils are located in the north-central part of the INEEL site. The colluvial sediments are located along the western edge of the site. Wind-blown sediments (silt and sand) covering lava plains occupy the rest of the site's landscape (LMITCO 1997). The thickness of surficial sediments ranges from less than 0.3 meters (1 foot) at basalt outcrops east of INTEC to 95 meters (313 feet) near the Big Lost River sinks (DOE 1999a). No prime farmland lies within the INEEL boundaries (DOE 1999h).

The nearest capable fault to ANL-W is the South Creek Segment of the Lemhi Fault, which is located 31 kilometers (19 miles) northwest of the site (Abbott, Crockett, and Moor 1997). ANL-W is located within the Axial Volcanic Zone, which has an estimated recurrence interval for volcanism of 16,000 years (Hackett and Smith 1994). The site is situated within a topographically closed basin. Low ridges of basalt found east of the area rise as high as 30 meters (100 feet) above the level of the plain. Sediments cover most of the underlying basalt on the plain, except where pressure ridges form basalt outcrops (ANL 1999a). Soils in the ANL-W area have been found to resemble the Pancheri-Polatis-Tenno series, which generally consists of light brown-gray well-drained silty loams to brown extremely stony loams (ANL 1998a, DOA 1973). Soils are highly disturbed within developed areas of the site.

3.2.6 Ecological Resources

Ecological resources include terrestrial resources, wetlands, aquatic resources, and threatened and endangered species. Material presented in this section, unless otherwise noted, is from the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement* (DOE 1996c).

3.2.6.1 Terrestrial Resources

INEEL lies in a cool desert ecosystem dominated by shrub-steppe communities. Most land within the site is relatively undisturbed and provides habitat for species native to the region. The importance of this habitat was recently recognized when approximately 29,950 hectares (74,000 acres) located in the north central part of the site were designated as the INEEL Sagebrush Steppe Ecosystem Reserve (Figure 3–1) (DOE 1999g). Facilities and operating areas occupy only 2 percent of INEEL. Although sagebrush communities occupy about 80 percent of INEEL, a total of 20 plant communities have been identified (**Figure 3–4**). The interspersion

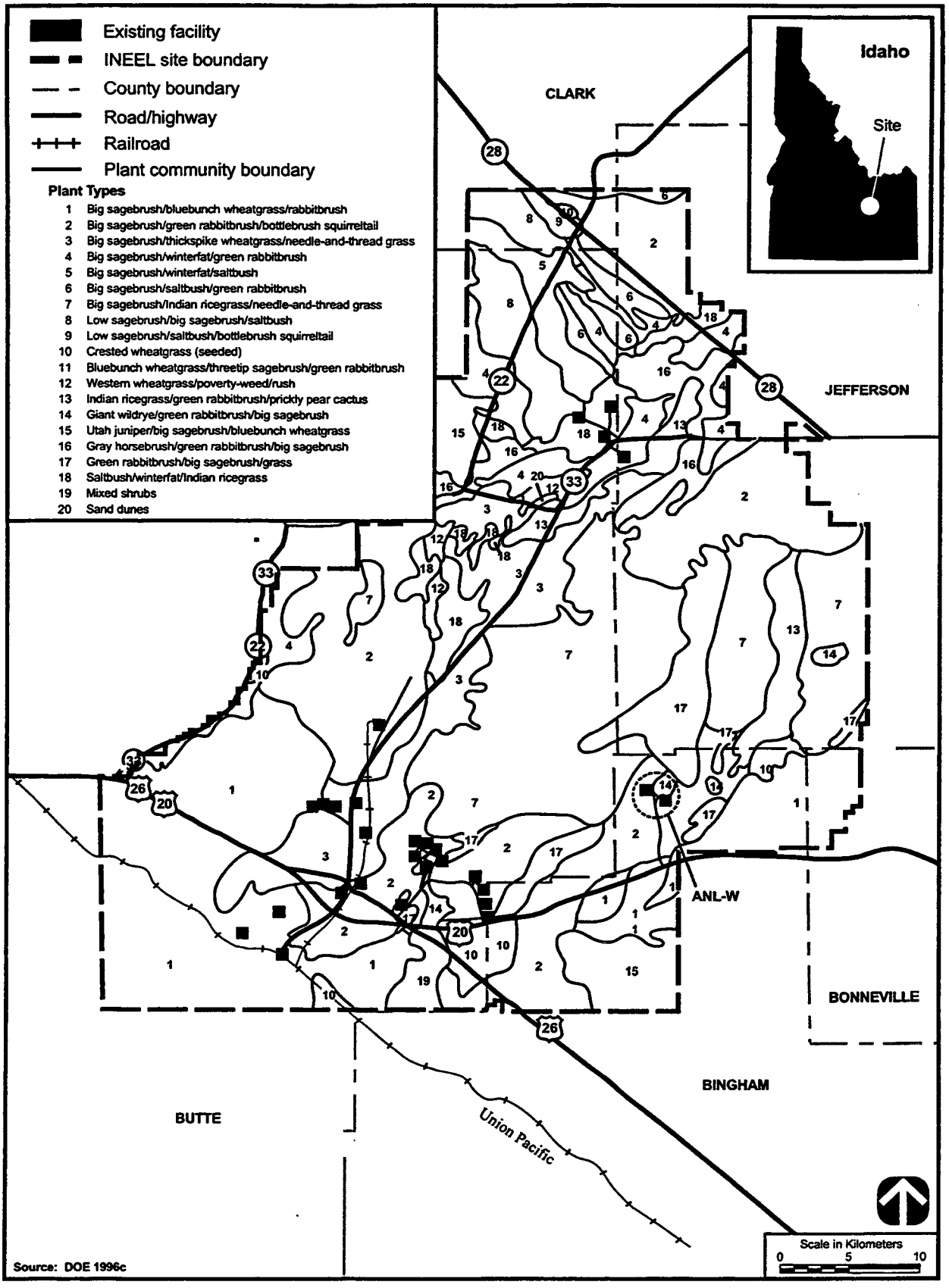

Figure 3–4 Distribution of Plant Communities at INEEL

of low and big sagebrush communities in the northern portion of INEEL and juniper communities located in the northwestern and southeastern portions of the site are considered sensitive habitats. The former provides critical winter and spring range for sage grouse and pronghorn, while the latter is important to nesting raptors and songbirds. Riparian vegetation, primarily cottonwood and willow, along the Big Lost River and Birch Creek also provides nesting habitat for hawks, owls, and songbirds. In total, 398 plant taxa have been documented on INEEL.

INEEL supports numerous animal species, including 2 amphibian, 11 reptile, 225 bird, and 44 mammal species (ESRF 1999). Common animals on the INEEL site include the short-horned lizard, gopher snake, sage sparrow, Townsend's ground squirrel, and black-tailed jackrabbit. Important game animals include the sage grouse, mule deer, elk, and pronghorn. During some winters, 4,500 to 6,000 pronghorn, or about 30 percent of Idaho's total population, may be found on the INEEL site. Pronghorn wintering areas are located in the northeastern portion of the site, in the area of the Big Lost River sinks, in the west-central portion of the site along the Big Lost River, and in the south-central portion of the site (DOE 1996c). Hunting of pronghorn and elk to control crop damage is permitted on site within 0.8 kilometers (0.5 miles) of the site boundary (LMITCO 1997). Numerous raptors, such as the golden eagle and prairie falcon, and carnivores, such as the coyote and mountain lion, also are found on the INEEL site.

ANL-W is located within one of several sagebrush communities found on the INEEL site (Figure 3–4). While sagebrush is present on undeveloped portions of the site, developed areas are nearly devoid of vegetation. Wildlife use of developed portions of the site is negligible; however, surrounding areas do provide natural habitat for a variety of wildlife. While elk and mule deer are the most important large mammals present in the area, many of the common species discussed above also would be expected. The ANL-W wastewater pond acts as an important source of water for wildlife found in the vicinity of the site (Cieminski and Flake 1995).

3.2.6.2 Wetlands

National Wetland Inventory maps prepared by the U.S. Fish and Wildlife Service indicate that the primary wetland areas on the INEEL site are associated with the Big Lost River, the Big Lost River spreading areas, and the Big Lost River sinks (or playas) (Figure 3–2). Smaller isolated wetlands (less than 0.4 hectares [1 acre]) also occur on the site (DOE 1996c). The only area of jurisdictional wetland is the Big Lost River sinks (Evans et al. 1998).

Wetland vegetation exists along the Big Lost River, which is located 18 kilometers (11 miles) west of ANL-W; however, this vegetation is in poor condition because of recent years of only intermittent flows. The Big Lost River spreading areas and Big Lost River sinks are seasonal wetlands and are located 34 kilometers (21 miles) west-southwest and 23 kilometers (14 miles) northwest of ANL-W, respectively. These areas can provide more than 809 hectares (2,000 acres) of wetland habitat during wet years. Within ANL-W itself, small areas of intermittent marsh occur along cooling tower blowdown ditches (Morris 1996).

3.2.6.3 Aquatic Resources

Aquatic habitat on the INEEL site is limited to the Big Lost River, Little Lost River, Birch Creek, and a number of liquid-waste disposal ponds. All three streams are intermittent and drain into four sinks in the north-central part of the site. Six species of fish have been observed within water bodies located on the site (ESRF 1999). Species observed in the Big Lost River include brook trout, rainbow trout, mountain whitefish, speckled dace, shorthead sculpin, and kokanee salmon. The Little Lost River and Birch Creek enter INEEL only during periods of high flow. Surveys of fish in these surface water bodies have not been conducted. The liquid waste disposal ponds on the INEEL site, while considered aquatic habitat, do not support fish.

There is no natural aquatic habitat on or in the vicinity of the ANL-W site. The nearest such habitat is the Big Lost River, which is located 18 kilometers (11 miles) west of the site. ANL-W waste disposal ponds do not contain any fish populations, but do provide habitat for a variety of aquatic invertebrates (Cieminski and Flake 1995).

3.2.6.4 Threatened and Endangered Species

Nineteen federally and state-listed threatened, endangered, and other special status species may be found on and in the vicinity of the INEEL site, 12 of which have been observed at the site (see Table 3–1 of the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement* [DOE 1996c]). Two of these species are federally and/or state-listed. The bald eagle is listed by the U.S. Fish and Wildlife Service as threatened and by the State of Idaho as endangered. The peregrine falcon is listed by the state as endangered. The bald eagle rarely has been seen in the western and northern portions of INEEL. The peregrine falcon is an infrequent visitor to the site. The occurrence of the gray wolf (listed endangered, experimental populations) on the INEEL site is unverified. No critical habitat for threatened or endangered species, as defined in the Endangered Species Act, exists on the INEEL site.

The ANL-W area was surveyed in 1996 for threatened, endangered, and special status species (Morris 1996). The only listed species observed were the peregrine falcon and the loggerhead shrike. While no peregrine falcon nests were found near ANL-W, one perigrene falcon was observed perched on a power line 1.5 kilometers (0.9 miles) from the site. The loggerhead shrike, which is listed by Idaho as a species of concern, has been seen on numerous occasions in the vicinity of the site. The gray wolf (state endangered) and the pigmy rabbit and Townsend's big-eared bat (state species of concern) were not identified in the vicinity of ANL-W during the surveys. In addition, no federally or state-listed plants were found in the vicinity of the site. Consultation has been conducted with both the U.S. Fish and Wildlife Service and the state.

3.2.7 Cultural and Paleontological Resources

Cultural resources are human imprints on the landscape and are defined and protected by a series of Federal laws, regulations, and guidelines. INEEL has a well-documented recording of cultural and paleontological resources. Guidance for the identification, evaluation, and management of these resources is included in the *Idaho National Engineering Laboratory Management Plan for Cultural Resources (Final Draft)* (Miller 1995). Past studies, which covered 4 percent of the site, identified 1,506 cultural resource sites and isolated finds, including 688 prehistoric sites, 38 historic sites, 753 prehistoric isolates, and 27 historic isolates (DOE 1996c). As of January 1998, approximately 7 percent of INEEL has been surveyed, raising the number of potentially significant archaeological sites to 1,839 (DOE 1999a). Most surveys have been conducted near major facility areas in conjunction with modification, demolition, or abandonment of site facilities.

3.2.7.1 Prehistoric Resources

Prehistoric resources are physical properties remaining from human activities that predate written records. Prehistoric resources identified at INEEL are generally reflective of Native American hunting and gathering activities. Resources appear to be concentrated along the Big Lost River and Birch Creek, atop buttes, and within craters or caves. They include residential bases, campsites, caves, hunting blinds, rock alignments, and limited-activity locations such as lithic and ceramic scatters, hearths, and concentrations of fire-affected rock. Most sites have not been evaluated formally for nomination to the National Register, but are considered to be potentially eligible. Given the rather high density of prehistoric sites at INEEL, additional sites are likely to be identified as surveys continue.

The most recent cultural resource survey conducted near ANL-W took place in 1996 and covered an area to the south of the site that had been burned over by a wildfire and was proposed for revegetation (CEEA 1996).

A total of 12 isolated finds and 2 archaeological sites were located. Isolated finds include items such as pieces of Shoshone brownware pottery and projectile points. The archaeological sites include projectile points, scrappers, and volcanic glass flakes. Areas within the fenced portion of ANL-W are highly disturbed and are not likely to yield significant archaeological material.

3.2.7.2 Historic Resources

Thirty-eight historic sites and 27 historic isolates have been identified at INEEL. These resources are representative of European-American activities, including fur trapping and trading, immigration, transportation, mining, agriculture, and homesteading, as well as more recent military and scientific/engineering research and development activities. Examples of historic resources include Goodale's Cutoff (a spur of the Oregon Trail), remnants of homesteads and ranches, irrigation canals, and a variety of structures from the World War II era. EBR-I, the first reactor to achieve a self-sustaining chain reaction using plutonium instead of uranium as the principal fuel component, is listed on the National Register and is designated a National Historic Landmark. Many other INEEL structures built between 1949 and 1974 are considered eligible for the National Register because of their exceptional scientific and engineering significance and their major role in the development of nuclear science and engineering since World War II. Additional historic sites are likely to exist in unsurveyed portions of INEEL.

A number of recent items including farm implements, a belt buckle, broken glass, and a large scattering of cans have been found in the vicinity of ANL-W (CEEA 1996). EBR-II has been designated as an American Nuclear Society Historical Landmark (DOE 1997c). Consultation has been conducted with the State Historic Preservation Office.

3.2.7.3 Native American Resources

Native American resources at INEEL are associated with the two groups of nomadic hunters and gatherers that used the region at the time of European-American contact: the Shoshone and Bannock. Both of these groups used the area that now encompasses INEEL as they harvested plant and animal resources and obsidian from Big Southern Butte or Howe Point. Because INEEL is considered part of the Shoshone-Bannock Tribes' ancestral homeland, it contains many localities that are important for traditional, cultural, educational, and religious reasons. This includes not only prehistoric archaeological sites, which are important in a religious or cultural heritage context, but also features of the natural landscape and air, plant, water, or animal resources that have special significance. The value of certain areas on the INEEL site was recognized in the *1994 Memorandum of Agreement with the Shoshone-Bannock Tribes* (DOE 1994a), which provides Tribal members access to the Middle Butte area to perform sacred or religious ceremonies or other educational or cultural activities.

Although prehistoric Native American resources have been found in the vicinity of ANL-W (see Prehistoric Resources), the 1994 Memorandum of Agreement with the Shoshone-Bannock Tribes (DOE 1994a) does not affect the site (DOE 1997c). Consultation has been conducted with the Shoshone and Bannock Tribes.

3.2.7.4 Paleontological Resources

Paleontological resources are the physical remains, impressions, or traces of plants or animals from a former geologic age. The region encompassing INEEL has abundant and varied paleontological resources, including plant, vertebrate, and invertebrate remains from soils; lake and river sediments; and organic materials found in caves and archaeological sites. Vertebrate fossils recovered from the Big Lost River flood plain consist of isolated bones or teeth from large mammals of the Pleistocene or Ice Age. Fossils have been recorded in the vicinity of the Naval Reactors Facility, and a single mammoth tooth was salvaged during the excavation of a percolation pond immediately south of INTEC. Occasional fossil mammoth, horse, and camel skeletal

elements have been retrieved from the Big Lost River diversion dam and the Radioactive Waste Management Complex on the southwestern side of INEEL, and from river and alluvial fan gravels and Lake Terreton sediments near Test Area North (Abbott, Crockett, and Moor 1997). In total, 24 paleontological localities have been identified at INEEL (Miller 1995).

Paleontological resources were not found in the immediate vicinity of ANL-W during a recent archaeological survey (CEEA 1996).

3.2.8 Socioeconomics

Statistics for employment and economy are presented for the regional economic area, which encompasses 13 counties around INEEL located in Idaho and Wyoming. Statistics for population and housing, community services, and local transportation are presented for the region of influence. The region of influence is a 4-county area in Idaho in which 94.4 percent of all INEEL employees reside (**Table 3–7**). In 1997, the total INEEL employment was 8,291 persons (5.5 percent of the regional economic area civilian labor force).

Table 3–7 Distribution of Employees by Place of Residence in the INEEL Region of Influence, 1997

County	*Number of Employees*	*Total Site Employment (Percent)*
Bonneville	5,553	67
Bingham	1,077	13
Bannock	615	7.4
Jefferson	583	7
Region of influence total	**7,828**	**94.4**

Source: DOE 1999h.

3.2.8.1 Regional Economy Characteristics

Between 1990 and 1996, the civilian labor force in the regional economic area increased 26 percent to the 1996 level of 150,403. In 1996, the annual unemployment average in the regional economic area was 4.8 percent, slightly less than the annual unemployment average for Idaho (5.2 percent) and Wyoming (5 percent).

In 1995, service activities represented the largest sector of employment in the regional economic area (27.1 percent). This was followed by retail trade (20.4 percent) and government (19.5 percent). The totals for these employment sectors in Idaho were 21.5 percent, 19.6 percent, and 18.7 percent, respectively. The totals for these employment sectors in Wyoming were 21.1 percent, 20.8 percent, and 25 percent, respectively.

3.2.8.2 Population and Housing

In 1996, the region of influence population totaled 213,547. Between 1990 and 1996 the region of influence population increased by 10.6 percent, compared with a 17.5 percent increase in Idaho's population. Between 1980 and 1990 the number of housing units in the region of influence increased by 6.7 percent, compared with a 10.2 percent increase in Idaho (DOE 1999h). The total number of housing units in the region of influence for 1990 was 69,760. In 1995, the total number of owner and renter housing units within the region of influence was 74,600 (DOE 1996a). The 1990 region of influence homeowner vacancy rate was 2.1 percent, compared with Idaho's rate of 2 percent. The region of influence renter vacancy rate was 8.3 percent, compared with Idaho's rate of 7.3 percent.

3.2.8.3 Community Services

Community services include public education and public safety. In 1997, school districts providing public education in the INEEL region of influence were operating at capacities of between 50 to 100 percent. Total student enrollment in the INEEL region of influence in 1997 was 50,168, and the student-to-teacher ratio averaged 18.8 to 1. In 1990, the average student-to-teacher ratio for Idaho was 12.8 to 1. In 1997, a total of 475 sworn police officers were serving the four-county region of influence. The average INEEL region of influence officer-to-population ratio was 2.2 officers per 1,000 persons. This compares with the 1990 state average of 1.5 officers per 1,000 persons.

3.2.8.4 Local Transportation

Vehicular access to INEEL is provided by U.S. Routes 20 and 26 to the south and State Routes 22 and 33 to the north. U.S. Routes 20 and 26 and State Routes 22 and 33 all share rights-of-way west of INEEL (Figure 3–1). DOE shuttle vans provide transportation between INEEL facilities and Idaho Falls for DOE and contractor personnel. The major railroad in the region of influence is the Union Pacific Railroad. The railroad's Blackfoot-to-Arco Branch provides rail service to the southern portion of INEEL. A DOE-owned spur connects the Union Pacific Railroad to INEEL by a junction at Scovill Siding. There are no navigable waterways within the region of influence capable of accommodating waterborne transportation of material shipments to INEEL. Fanning Field in Idaho Falls and Pocatello Municipal Airport in Pocatello provide jet air passenger and cargo service for both national and local carriers.

3.2.9 Environmental Justice

Under Executive Order 12898, *Federal Actions to Address Environmental Justice in Minority Populations and Low-Income Populations*, Federal agencies are responsible for identifying and addressing the possibility of disproportionately high and adverse health or environmental effects of programs and policies on minority and low-income populations in potentially affected areas. Minority populations refer to all people of color, exclusive of white non-Hispanics. Low-income populations refer to households whose incomes are below the Federal poverty threshold. In the case of INEEL, the potentially affected area includes only parts of central Idaho.

The 1990 census data show that the percentage of minorities within the contiguous United States was 24.1 percent, while within the State of Idaho it was 7.7 percent. The data also show that 13.1 percent of the incomes within the United States were below the poverty threshold. Within Idaho, 13.3 percent of the incomes were below the poverty threshold.

The potentially affected area surrounding ANL-W is defined by a circle with an 80-kilometer (50-mile) radius centered at latitude 43°35'41.7" N, longitude 112°39'18.7" W. The total population residing within that area in 1990 was 180,582. The proportion of this population that was considered minority was 8.7 percent. At the time of the 1990 census, Hispanics and Native Americans were the largest minority groups within that area, constituting 5.2 percent and 2.2 percent of the total population, respectively. Asians constituted about 1 percent and Blacks about 0.3 percent.

A breakdown of incomes in the potentially affected area also is available from the 1990 census data. At that time, the poverty threshold was $9,981 for a family of three with one related child under 18 years of age. A total of 25,046 persons (15 percent of the total population) residing within the potentially affected area around ANL-W reported incomes below that threshold.

3.2.10 Existing Human Health Risk

Public and occupational health and safety issues include the determination of potentially adverse effects on human health that result from acute and chronic exposures to ionizing radiation and hazardous chemicals.

3.2.10.1 Radiation Exposure and Risk

Major sources and levels of background radiation exposure to individuals in the vicinity of INEEL are shown in **Table 3–8**. Annual background radiation doses to individuals are expected to remain constant over time. The total dose to the population, in terms of person-rem, changes as the population size changes. Background radiation doses are unrelated to INEEL operations.

Table 3–8 Sources of Radiation Exposure to Individuals in the INEEL Vicinity Unrelated to INEEL Operations

Source	*Effective Dose Equivalent (millirem per year)*
Natural Background Radiation [a]	
Cosmic radiation	48
External terrestrial radiation	74
Internal terrestrial/cosmogenic radiation	40
Radon in homes (inhaled)	200 [b,c]
Other Background Radiation [c]	
Diagnostic x-rays and nuclear medicine	53
Weapons test fallout	less than 1
Air travel	1
Consumer and industrial products	10
Total	**427**

[a] Evans et al. 1998.
[b] An average for the United States.
[c] NCRP 1987.

Releases of radionuclides to the environment from INEEL operations provide another source of radiation exposure to individuals in the vicinity of INEEL. Types and quantities of radionuclides released from INEEL operations in 1997 are listed in the *Idaho National Engineering Laboratory Site Environmental Report for Calendar Year 1997* (Evans et al. 1998). The doses to the public resulting from these releases are presented in **Table 3–9**. These doses fall within radiological limits per DOE Order 5400.5, *Radiation Protection of the Public and the Environment*, and are much lower than those of background radiation.

Using a risk estimator of 500 latent cancer deaths per 1 million person-rem to the public (see Appendix E), the fatal cancer risk to the maximally exposed member of the public due to radiological releases from INEEL operations in 1997 is estimated to be 1.1×10^{-8}. That is, the estimated probability of this person dying of cancer at some point in the future from radiation exposure associated with one year of INEEL operations is less than 2 in 100 million. (It takes several to many years from the time of radiation exposure for a cancer to manifest itself.)

Table 3–9 Radiation Doses to the Public From Normal INEEL Operations in 1997 (Total Effective Dose Equivalent)

Members of the Public	Atmospheric Releases		Liquid Releases		Total	
	Standard [a]	Actual	Standard [a]	Actual	Standard [a]	Actual
Maximally exposed offsite individual (millirem)	10	0.021	4	0	100	0.021
Population within 80 kilometers (50 miles) (person-rem) [b]	None	0.23	None	0	100	0.23
Average individual within 80 kilometers (50 miles) (millirem) [c]	None	0.0019	None	0	None	0.0019

[a] The standards for individuals are given in DOE Order 5400.5. As discussed in that Order, the 10-millirem per year limit from airborne emissions is required by the Clean Air Act, and the 4-millirem per year limit is required by the Safe Drinking Water Act. For this EIS, the 4-millirem per year value is assumed conservatively to be the limit for the sum of doses from all liquid pathways. The total dose of 100 millirem per year is the limit from all pathways combined. The 100-person-rem value for the population is given in the proposed 10 CFR 834, *Radiation Protection of the Public and Environment; Proposed Rule,* as published in 58 FR 16268. If the potential total dose exceeds the 100 person-rem value, the contractor operating the facility is required to notify DOE.

[b] About 121,400 in 1997.

[c] Obtained by dividing the population dose by the number of people living within 80 kilometers (50 miles) of the site.

Source: Evans et al. 1998.

According to the same risk estimator, 0.0012 excess latent fatal cancers are projected in the population living within 80 kilometers (50 miles) of INEEL from normal operations in 1997. To place this number in perspective, it may be compared with the number of fatal cancers expected in the same population from all causes. The 1995 mortality rate associated with cancer for the entire U.S. population was 0.2 percent per year. Based on this mortality rate, the number of latent fatal cancers expected during 1997 from all causes in the population living within 80 kilometers (50 miles) of INEEL was 243. This expected number of fatal cancers is much higher than the 0.00012 fatal cancers estimated from INEEL operations in 1997.

INEEL workers receive the same dose as the general public from background radiation, but they also receive an additional dose from working in facilities with nuclear materials. The average dose to the individual worker and the cumulative dose to all workers at INEEL from operations in 1997 are presented in **Table 3–10**. These doses fall within the radiological regulatory limits of 10 CFR 835 (DOE 1995a). According to a risk estimator of 400 latent fatal cancers per 1 million person-rem among workers (see Appendix E), the number of projected fatal cancers among INEEL workers from normal operations in 1997 is 0.046. The risk estimator for workers is lower than the estimator for the public because of the absence from the work force of the more radiosensitive infant and child age groups.

Table 3–10 Radiation Doses to Workers From Normal INEEL Operations in 1997 (Total Effective Dose Equivalent)

Occupational Personnel	Onsite Releases and Direct Radiation	
	Standard [a]	Actual
Average radiation worker (millirem)	None [b]	101 [c]
Total workers (person-rem) [d]	None	115 [c]

[a] The radiological limit for an individual worker is 5,000 millirem per year. However, DOE's goal is to maintain radiological exposure as low as is reasonably achievable. Therefore, DOE has established an administrative control level of 2,000 millirem per year (DOE Order N 441.1); the site must make reasonable attempts to maintain individual worker doses below this level.

[b] No standard is specified for an "average radiation worker"; however, the maximum dose that this worker may receive is limited to that given in footnote "a."

[c] Does not include doses received at the Naval Reactors Facility. The impacts associated with this facility fall under the jurisdiction of the Navy as part of the Nuclear Propulsion Program.

[d] 1,141 workers with measurable doses in 1997.

Sources: DOE 1995a, DOE 1998d.

A more detailed presentation of the radiation environment, including background exposures and radiological releases and doses, is presented in the *Idaho National Engineering and Environmental Laboratory Site Environmental Report for Calendar Year 1997* (Evans et al. 1998). The concentrations of radioactivity in various environmental media (including air, water, and soil) in the site region (on and off the site) also are presented in that report.

External radiation doses and concentrations of plutonium in air have been measured at ANL-W. The onsite dose is measured for comparison against natural background levels measured at offsite control locations; the numerical differences in these measurements may be directly attributable to radiological sources that are located in the vicinity of the onsite measurement location. In 1997, the annual average dose within the area was about 144 millirem. This is about 5 millirem higher than the average dose measured at offsite control locations. Concentrations in air of plutonium-239 and plutonium-240 in 1996 were 3.4×10^{-18} microcuries per milliliter. This value is essentially the same as those measured at an offsite control location. Finally, concentrations in air of gross alpha and beta radiation at ANL-W are 6.0×10^{-16} microcuries per milliliter and 2.0×10^{-14} microcuries per milliliter, respectively. These alpha and beta radiation concentrations are essentially the same as those measured at offsite control locations (Evans et al. 1998).

3.2.10.2 Chemical Environment

Table 3–2 identifies the hazardous (i.e., carcinogenic and toxic/noncarcinogenic) chemicals that are emitted to the air at INEEL. The list includes only those chemicals that have ambient air quality standards and would be emitted under any one of the alternatives analyzed in this EIS. These include 1,3-butadiene, acetaldehyde, acrolein, benzene, cadmium, formaldehyde, toluene, and xylene.

Health impacts on the public may occur by inhaling air containing hazardous chemicals, ingesting contaminated drinking water or food, and direct exposure (skin contact). The primary health impacts from exposure to hazardous chemicals are from inhalation. Two major health effects are observed from the listed chemicals, the carcinogenic effect and the noncarcinogenic effect. These are presented below.

Carcinogenic Effects: These effects are estimated as the incremental probability of an individual developing cancer over a lifetime as a result of exposure to the potential carcinogen. This could be incremental or excess individual lifetime cancer risk.

Noncarcinogenic Effects: These effects are determined by the ratio between the calculated, or measured, concentration of the chemical in the air and the reference concentration or dose. This ratio is known as the hazard quotient. Hazard quotients for noncarcinogens are summed to obtain the hazard index. If the hazard index is less than 1, then no adverse health effects are expected.

For some chemicals where the weight of evidence is weak and carcinogenecity is not well established, the impacts of both cancer and noncancer effects are determined. **Table 3–11** summarizes the baseline hazardous chemical impacts to the public. This table lists only those chemicals for which reference concentrations for cancer or toxicity are available from the Integrated Risk Information System. The baseline concentrations are estimates of the highest existing concentrations and represent the highest concentrations to which individuals from the public could be exposed under normal operations (excluding accident conditions). These concentrations are in compliance with applicable guidelines. Additional information on estimating the health impacts of hazardous chemicals is presented in Appendix E, Section E.5.

The exposure of workers to hazardous chemicals varies among facilities and the operational activities, and the available information is insufficient for a meaningful estimate of impacts. Workers are protected by adherence to the Occupational Health and Safety Administration (OSHA) and EPA standards that regulate workplace atmospheric and drinking water concentrations of potentially hazardous chemicals. Monitoring the frequency

and the amount of chemicals released in operational processes ensures that these standards are not exceeded. Further, DOE requires that the environment in the workplace be as free as possible from recognized hazards that cause, or are likely to cause, illness or physical harm. Therefore, workplace conditions at INEEL are substantially better than required by standards.

Table 3–11 Hazardous Chemical Impacts to the Public from Existing Activities at INEEL

Chemical	*Hazard Quotient*	*Cancer Risk*
1,3-Butadiene	Not applicable	2.8×10^{-7}
Acetaldehyde	0.0002	2.4×10^{-8}
Acrolein	0.01340	Not applicable
Cadmium	Not applicable	7.5×10^{-8}
Benzene	Not applicable	2.4×10^{-7}
Formaldehyde	Not applicable	1.6×10^{-7}
Toluene	Less than 0.01	Not applicable
Hazard Index	Less than 0.0236	Not applicable

Source: DOE 1995a, or dispersion modeling.

3.2.10.3 Health Effects Studies

Epidemiological studies were conducted on communities surrounding INEEL to determine whether there are excess cancers in the general population. Two of these are described in more detail in Appendix M.4.4 of the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement* (DOE 1996c). No excess cancer mortality was reported, and although excess cancer incidence was observed, no association with INEEL was established. A study by the State of Idaho completed in June 1996 found excess brain cancer incidence in the six counties surrounding INEEL, but a follow-up survey concluded that, "There was nothing that clearly linked all these cases to one another or any one thing" (DOE 1996c).

No occupational epidemiological studies have been completed at INEEL to date, but several worker health studies were initiated recently at INEEL and another is almost complete. Researchers from the Boston University School of Public Health, in cooperation with the National Institute of Occupational Safety and Health, are investigating the effects of work force restructuring (downsizing) in the nuclear weapons industry. The health of displaced workers will be studied. Under a National Institute of Occupational Safety and Health cooperative agreement, the epidemiologic evaluation of childhood leukemia and paternal exposure to ionizing radiation now includes INEEL as well as other DOE sites. Another study begun in October 1997, *Medical Surveillance for Former Workers at INEEL*, is being carried out by a group of investigators consisting of the Oil, Chemical, and Atomic Workers International Union; Mount Sinai School of Medicine; the University of Massachusetts at Lowell; and Alice Hamilton College. A mortality study of the work force at INEEL being conducted by National Institute of Occupational Safety and Health is pending publication. DOE has implemented an epidemiologic surveillance program to monitor the health of current INEEL workers. A discussion of this program is given in Appendix M.4.4 of the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement* (DOE 1996c).

3.2.10.4 Accident History

DOE conducted a study, the *Idaho National Engineering Laboratory Historical Dose Evaluation,* to estimate the potential offsite radiation doses for the entire operating history of INEEL (DOE 1996c). Releases resulted from a variety of tests and experiments as well as a few accidents at INEEL. The study concluded that these releases contributed to the total radiation dose during test programs of the 1950s and early 1960s. The frequency and size of releases has declined since that time. There have been no serious unplanned or

accidental releases of radioactivity or other hazardous substances at INEEL facilities in the last 10 years of operation.

3.2.10.5 Emergency Preparedness

Each DOE site has established an emergency management program that would be activated in the event of an accident. This program has been developed and maintained to ensure adequate response to most accident conditions and to provide response efforts for accidents not specifically considered. The emergency management program includes emergency planning, training, preparedness, and response.

Government agencies whose plans are interrelated with the INEEL Emergency Plan for Action include the State of Idaho; Bingham, Bonneville, Butte, Clark, and Jefferson counties; the Bureau of Indian Affairs; and the Fort Hall Indian Reservation. INEEL contractors are responsible for responding to emergencies at their facilities. Specifically, the emergency action director is responsible for recognition, classification, notification, and protective action recommendations. At INEEL, emergency preparedness resources include fire protection from onsite and offsite locations and radiological and hazardous chemical material response. Emergency response facilities include an emergency control center at each facility, at the INEEL warning communication center, and at the INEEL site emergency operations center. Seven INEEL medical facilities are available to provide routine and emergency service. In addition, DOE has specified actions to be taken at all DOE sites to implement lessons learned from the emergency response to an accidental explosion at Hanford in May 1997.

3.2.11 Waste Management

Waste management includes minimization, characterization, treatment, storage, transportation, and disposal of waste generated from ongoing DOE activities. The waste is managed using appropriate treatment, storage, and disposal technologies, and is in compliance with all applicable Federal and state statutes and DOE Orders.

3.2.11.1 Waste Inventories and Activities

INEEL manages the following types of waste: high-level radioactive, transuranic, mixed transuranic, low-level radioactive, mixed, hazardous, and nonhazardous. Waste generation rates and the inventory of stored waste from activities at INEEL are provided in **Table 3–12**. The INEEL waste management capabilities are summarized in **Table 3–13**. More detailed descriptions of the waste management system capabilities at INEEL are included in the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement* (DOE 1996c) and the Programmatic Spent Nuclear Fuel EIS (DOE 1995a).

The EPA placed INEEL on the National Priorities List[3] on December 21, 1989. In accordance with CERCLA, DOE entered into a Consent Order with the EPA and the State of Idaho to coordinate cleanup activities at INEEL under one comprehensive strategy. This agreement integrates DOE's CERCLA response obligations with RCRA corrective action obligations. Aggressive plans are in place to achieve early remediation of sites that represent the greatest risk to workers and the public. The goal is to complete remediation of contaminated sites at INEEL to support delisting from the National Priorities List by the year 2019 (DOE 1996c). More information on regulatory requirements for waste disposal is provided in Chapter 5.

[3]*The National Priorities List is a list of those sites requiring cleanup that appear to have the most serious threat to public health or the environment due to the release of hazardous substances. The list is promulgated by the EPA under CERCLA.*

Table 3–12 Waste Generation Rates and Inventories at INEEL

Waste Type	Generation Rate (cubic meters per year)	Inventory (cubic meters)
High-level radioactive	0 [a]	4,000 [b]
Transuranic	0 [a, c]	65,000 [d]
Low-level radioactive	6,400 [e]	6,000 [f]
Mixed [g]	230	1,700
Hazardous	835 [c, h]	Not applicable [i]
Nonhazardous		
Liquid	2,000,000 [c, j]	Not applicable [i]
Solid	62,000 [c]	Not applicable [i]

[a] Refer to the text.
[b] INEEL 1999b. The inventory is calcined high-level radioactive waste.
[c] Moor and Peterson 1999.
[d] DOE 1995a.
[e] LMITCO 1998.
[f] Bright 1999.
[g] DOE 1998c.
[h] Includes 760 cubic meters that are recyclable.
[i] Generally, hazardous and nonhazardous waste is not held in long-term storage.
[j] Projected annual average generation amounts for 1997 to 2006.

Note: To convert from cubic meters to cubic yards, multiply by 1.31.
Sources: Given in footnotes b through g, above.

Table 3–13 Waste Management Capabilities at INEEL

Facility Name/Description	Capacity	Status	Applicable Waste Type						
			HLW	TRU	Mixed TRU	LLW	Mixed	Haz	Non-Haz
Treatment Facility (cubic meters per year except as otherwise specified)									
INTEC High-Efficiency Particulate Air Filter Leach, cubic meters per day	0.21	On-line			X		X		
INTEC Debris Treatment and Containment, cubic meters per day	88	Waiting for Part B Permit			X		X		
Advanced Mixed Waste Treatment Project	6,500	Planned for 2003			X		X		
INTEC New Waste Calcining Facility	248	On-line	X		X				
ANL–W Remote Treatment Facility	42	Planned for 2000		X	X	X	X		
ANL–W Hot Fuel Examination Facility Waste Characterization Area	37	On-line		X	X				
INTEC Waste Immobilization Facility	48	Planned for 2020			X	X	X		
INTEC Liquid Effluent Treatment and Disposal Facility	11,365	On-line					X		
INTEC High-Level Radioactive Waste Evaporator	6,138	On-line			X	X	X		
INTEC Process Equipment Waste Evaporator	13,000	On-line			X	X	X		

Facility Name/Description	Capacity	Status	Applicable Waste Type						
			HLW	TRU	Mixed TRU	LLW	Mixed	Haz	Non-Haz
ANL–W Sodium Processing Facility	698	On-line					X		
Test Area North Cask Dismantlement	11	On-line					X		
Waste Reduction Operations Complex - Debris Sizing, kilograms per hour	1,149	Planned for 2000				X	X		
Waste Reduction Operations Complex - Macroencapsulation, kilograms per hour	2,257	Planned for 1999					X		
Waste Reduction Operations Complex - Stabilization, cubic meters per day	7.6	On-line					X		
Waste Experimental Reduction Facility	49,610	On-line				X	X	X	
INTEC Sewage Treatment Plant	3,200,000	On-line							X
Storage Facility (cubic meters)									
INTEC Tank Farm	12,533	On-line	X [a]		X		X		
INTEC Calcine Bin Sets	6,950	On-line	X						
ANL–W Radioactive Sodium Storage	75	On-line			X		X		
ANL–W Sodium Components Maintenance Shop	200	On-line					X		
ANL–W Radioactive Scrap and Waste Storage	193	On-line		X	X	X	X		
ANL–W EBR-II Sodium Boiler Drain Tank	64	On-line					X		
ANL–W Hot Fuel Examination Facility Waste Characterization Area	37	On-line		X	X				
INTEC Fluorinel Dissolution Process High-Efficiency Particulate Air Filter Storage	25	On-line			X		X		
INTEC New Waste Calcining Facility High-Efficiency Particulate Air Filter Storage	56	On-line			X		X		
INTEC Chemical Processing Plant-1619 Storage	45	On-line					X	X	
INTEC Chemical Processing Plant-1617 Staging	8,523	On-line					X	X	
Radioactive Waste Management Complex Transuranic Storage Area [b]	64,900	On-line		X	X	X	X		
Radioactive Waste Management Complex Waste Storage [b]	112,400	On-line		X	X	X	X		
Radioactive Waste Management Complex Intermediate-Level Storage	100	On-line		X					
Waste Reduction Operations Complex Power Burst Facility Mixed Waste Storage	129	On-line					X	X	

Facility Name/Description	Capacity	Status	Applicable Waste Type						
			HLW	TRU	Mixed TRU	LLW	Mixed	Haz	Non-Haz
Portable storage at Special Power Excursion Reactor Test IV	237	On-line					X	X	
Power Burst Facility Waste Experimental Reduction Facility Waste Storage Building	685	On-line					X	X	
Test Area North 647 Waste Storage	104	On-line					X	X	
Test Area North 628 Specific Manufacturing Complex Container Storage	125	On-line					X	X	
Disposal Facility (cubic meters per year)									
Radioactive Waste Management Complex Disposal Facility	37,700	On-line				X			
Central Facilities Area Landfill Complex	48,000	On-line							X
Percolation ponds	2,000,000	On-line							X

HAZ = hazardous waste, HLW = high-level radioactive waste, LLW = low-level radioactive waste, TRU = transuranic waste

[a] Sodium-bearing waste.

[b] For these facilities, the low-level radioactive waste and mixed waste are considered alpha-contaminated low-level radioactive waste and alpha-contaminated mixed waste (waste containing between 10 and 100 nanocuries of alpha activity per gram).

Sources: DOE 1999f, DOE 1999h.

3.2.11.2 High-Level Radioactive Waste

High-level radioactive waste at INEEL was generated in the process of extracting useful isotopes from spent nuclear fuel at INTEC. Most of this fuel was from the Naval Reactors Program. Most aqueous solutions from spent nuclear fuel processing and isotope extraction were concentrated by evaporation and separated into low- and high-level radioactive waste streams in the Process Equipment Waste Evaporator. The liquid high-level radioactive waste was stored in subsurface tanks and then transformed by calcination into solid metal oxides in a granular form. This calcination was completed in February 1998. The calcine is stored in stainless steel bins in near-surface concrete vaults where it awaits further processing into a form suitable for emplacement in a Federal repository. INEEL will meet the requirements of a December 1991 Consent Order with the State of Idaho and the EPA to cease the use of existing storage tanks without constructing new tanks. Subsequently, the calcined waste will be treated to meet RCRA provisions on a schedule to be negotiated with the State of Idaho under the Federal Facilities Compliance Act.

Although sodium-bearing waste is not high-level radioactive waste as specified in the Nuclear Waste Policy Act of 1982, it has been managed historically as high-level radioactive waste at INEEL. This is because some of the physical and chemical properties of these two waste types are similar, e.g., both are acidic and both contain similar radionuclides, including transuranics (DOE 1999i). About 5,300 cubic meters (1.4×10^6 gallons) of liquid sodium-bearing waste remain in the INTEC Tank Farm. It is anticipated that this waste will be calcined in the New Waste Calcining Facility and then be treated to meet RCRA provisions.

3.2.11.3 Transuranic Waste

Transuranic waste generated since 1972 is segregated into contact-handled and remote-handled categories and stored at the Radioactive Waste Management Complex in a form designed for eventual retrieval (DOE 1996c).

Some transuranic waste also is stored at the Radioactive Scrap and Waste Facility at ANL–W (DOE 1995a). There is very little transuranic waste generated at INEEL. Most of the transuranic waste in storage was received from the Rocky Flats Environmental Technology Site (DOE 1996a). Transuranic waste currently is being stored pending shipment to the Waste Isolation Pilot Plant. The first shipment of transuranic waste from INEEL was received at the Waste Isolation Pilot Plant on April 28, 1999 (DOE 1999d). Transuranic waste is treated to meet the Waste Isolation Pilot Plant Waste Acceptance Criteria, packaged in accordance with DOE and U.S. Department of Transportation requirements, and transported to the Waste Isolation Pilot Plant for disposal (DOE 1996c).

The existing treatment facilities for transuranic waste at INEEL are limited to testing, characterization, and repackaging. The planned Waste Characterization Facility will characterize (identify) transuranic waste and either reclassify it (if it is found to be low-level radioactive waste) for disposal on the site, or prepare it so that it meets Waste Isolation Pilot Plant Waste Acceptance Criteria (DOE 1996c).

The Advanced Mixed Waste Treatment Project will be operated as a private sector treatment facility after its construction is completed. This facility will: (1) treat waste to meet Waste Isolation Pilot Plant Waste Acceptance Criteria, RCRA Land Disposal Restrictions, and required Toxic Substances Control Act standards; (2) reduce waste volume and life cycle cost to DOE; and (3) perform tasks in a safe and environmentally compliant manner (INEEL 1999a).

Waste containing between 10 and 100 nanocuries of alpha activity per gram of transuranic radionuclides is called alpha low-level radioactive waste. Although this waste technically is considered low-level radioactive waste rather than transuranic waste, it cannot be disposed of at INEEL because it does not meet all INEEL low-level radioactive waste disposal facility acceptance criteria. Alpha low-level radioactive waste and alpha mixed waste are managed together as part of the Transuranic Waste program. It is expected that this waste will be treated by the Advanced Mixed Waste Treatment Project and then disposed of at the Waste Isolation Pilot Plant (DOE 1999h).

3.2.11.4 Low-Level Radioactive Waste

Liquid low-level radioactive waste either is evaporated and processed to a calcine form or solidified before disposal (DOE 1996a). INTEC has the capability to treat aqueous low-level radioactive waste. Liquid low-level radioactive waste is concentrated at the INTEC Process Equipment Waste Evaporator, and the condensed vapor is processed by the Liquid Effluent Treatment and Disposal Facility. The concentrated materials remaining after evaporation are pumped to the INTEC Tank Farm. Some small volumes of liquid low-level radioactive waste are solidified at the Waste Experimental Reduction Facility for disposal at the Radioactive Waste Management Complex. In addition, small volumes of aqueous low-level radioactive waste are discharged to the double-lined pond at the Test Reactor Area for evaporation (DOE 1995a).

Most solid low-level radioactive waste at INEEL is sent to the Waste Experimental Reduction Facility for treatment by incineration, compaction, size reduction, or stabilization before shipment for disposal at the Radioactive Waste Management Complex or offsite disposal facilities (DOE 1999h). Disposal occurs in pits and concrete-lined soil vaults in the subsurface disposal area of the Radioactive Waste Management Complex (DOE 1995a). About 40 percent of the low-level radioactive waste generated at INEEL (containing less than 10 nanocuries per gram of radioactivity) is buried in shallow trenches; the remaining 60 percent is buried at the Radioactive Waste Management Complex following treatment for volume reduction. Additionally, some low-level radioactive waste is shipped off site to be incinerated, and the residual ash is returned to INEEL for disposal. The Radioactive Waste Management Complex is expected to be filled to capacity by the year 2030, although some proposals would close the Low-Level Radioactive Waste Disposal Facility by 2006 (DOE 1999h).

3.2.11.5 Mixed Waste

Mixed waste is divided into two categories for management purposes: alpha mixed waste and beta-gamma mixed waste. Most of the alpha mixed waste stored at INEEL is waste that has been reclassified from mixed transuranic waste and is managed as part of the transuranic waste program. Therefore, this section deals only with beta-gamma mixed waste (DOE 1995a).

Mixed waste, including polychlorinated biphenyls–contaminated low-level radioactive waste, is stored in several onsite areas awaiting the development of treatment methods (DOE 1996c). Mixed waste is stored at the mixed waste storage facility (the Waste Experimental Reduction Facility Waste Storage Building) and in portable storage units at the Power Burst Facility area. In addition, smaller quantities of mixed waste are stored in various facilities at INEEL, including the Hazardous Chemical/Radioactive Waste Facility at INTEC and the Radioactive Sodium Storage Facility and Radioactive Scrap and Waste Storage Facility at ANL-W (DOE 1995a). Although mixed waste is stored in many locations at INEEL, the bulk of that volume is solid waste stored at the Radioactive Waste Management Complex (DOE 1996c).

Aqueous mixed waste is concentrated at INTEC. The condensate from the waste evaporator is processed by the Liquid Effluent Treatment and Disposal Facility. The concentrated material remaining after evaporation (mixed waste) is pumped to the INTEC Tank Farm for storage (DOE 1999h).

As part of the site treatment plans required by the Federal Facilities Compliance Agreement, preferred treatment options have been identified to eliminate the hazardous waste component for many types of mixed waste (DOE 1995a). Mixed waste is being or will be processed to RCRA Land Disposal Restrictions treatment standards through several treatment facilities. Those treatment facilities and their operational status are: (1) Waste Experimental Reduction Facility Incinerator (operational); (2) Waste Experimental Reduction Facility Stabilization (operational); (3) Test Area North Cask Dismantlement (operational); (4) Sodium Process Facility (operational); (5) High-Efficiency Particulate Air Filter Leach (operational); (6) Waste Reduction Operations Complex Macroencapsulation (June 2000); (7) Waste Reduction Operations Complex Mercury Retort (June 2000); (8) Debris Treatment (September 2000); and (9) Advanced Mixed Waste Treatment Project (March 2003). Commercial treatment facilities also are being considered, as appropriate. Currently, limited amounts of mixed waste are disposed of at Envirocare of Utah (DOE 1999h).

3.2.11.6 Hazardous Waste

Approximately 1 percent of the total waste generated at INEEL is hazardous waste. Most of the hazardous waste generated annually at INEEL is transported off site for treatment and disposal (DOE 1995a). Offsite shipments are surveyed to determine that the waste has no radioactive content and therefore are not mixed waste (DOE 1996c). Highly reactive or unstable materials, such as waste explosives, are addressed on a case-by-case basis, and are either stored, burned, or detonated, as appropriate.

3.2.11.7 Nonhazardous Waste

Approximately 90 percent of the waste generated at INEEL is classified as industrial waste and is disposed of on site in a landfill complex in the Central Facilities Area and off site at the Bonneville County landfill (DOE 1995a). The onsite landfill complex contains separate areas for petroleum-contaminated media, industrial waste, and asbestos waste (DOE 1999h). The onsite landfill is 5 hectares (12 acres), and is being expanded by 91 hectares (225 acres) to provide capacity for at least 30 years (DOE 1996c).

Sewage is disposed of in surface impoundments in accordance with the terms of the October 7, 1992, Consent Order. Wastewater in the impoundments is allowed to evaporate, and the resulting sludge is placed in the landfill. Solids are separated and reclaimed where possible (DOE 1996c). Nonhazardous service wastewater

generated at INTEC is disposed of in percolation ponds at a flow rate of 3.8 million to 7.6 million liters (1 million to 2 million gallons) per day. The INTEC sanitary sewer system collects and transfers sanitary waste to the sewage treatment lagoons east of INTEC for treatment and disposal. This system has a capacity of 3,200,000 cubic meters (4,190,000 cubic yards) per year (DOE 1999h).

3.2.11.8 Waste Minimization

The DOE Idaho Operations Office has an active waste minimization and pollution prevention program to reduce the total amount of waste generated and disposed of at INEEL. This is accomplished by eliminating waste through source reduction or material substitution; by recycling potential waste materials that cannot be minimized or eliminated; and by treating all of the waste that is generated to reduce its volume, toxicity, or mobility prior to storage or disposal. The Idaho Operations Office published its first waste minimization plan in 1990, which defined specific goals, methodology, responsibility, and achievements of programs and organizations. The achievements and progress have been updated at least annually. Implementation of pollution prevention projects reduced the total amount of waste generated at INEEL in 1997 by approximately 3,100 cubic meters (4,000 cubic yards) (DOE 1998b).

The INEEL waste minimization program has reduced significantly the quantities of hazardous waste generated at INEEL. For example, in 1992, 760 cubic meters (994 cubic yards) of hazardous waste were recycled. Recyclable hazardous materials include metals such as bulk lead, mercury, and chromium; solvents; fuel; and other waste materials (DOE 1995a). Soon the use of nonhazardous chemicals and the recycling of those for which there is no substitute should nearly eliminate the generation of hazardous waste (DOE 1996c).

Another goal of the INEEL waste minimization program is to reduce nonhazardous waste generation by 33 percent by the end of 1999 (DOE 1998b). During 1993 through 1995, INEEL recycled more than 680,400 kilograms (1.5 million pounds) of paper and cardboard (DOE 1999h). Efforts are also underway to expand the recycling program to include asphalt and metals and to convert scrap wood into mulch (DOE 1995a).

3.2.11.9 Preferred Waste Management Alternatives From the Final Waste Management Programmatic EIS and Associated Records of Decision

Preferred waste management alternatives from the *Final Waste Management Programmatic Environmental Impact Statement for Managing Treatment, Storage, and Disposal of Radioactive and Hazardous Waste* (Waste Management Programmatic EIS) (DOE 1997a) are shown in **Table 3–14** for the waste types analyzed in this EIS. Management of this waste could result in the construction of new waste management facilities at INEEL and the closure of other facilities. Decisions on the various waste types were announced in a series of Records of Decision that have been issued on the Waste Management Programmatic EIS. The transuranic waste Record of Decision was issued on January 20, 1998 (63 FR 3629); the hazardous waste Record of Decision on August 5, 1998 (63 FR 41810); the high-level radioactive waste Record of Decision on August 26, 1999 (64 FR 46661); and the low-level and mixed low-level radioactive waste Record of Decision on February 25, 2000 (65 FR 10061). The transuranic waste Record of Decision states, "...each of the Department's sites that currently has or will generate [sic] transuranic waste will prepare and store its transuranic waste on site...." The hazardous waste Record of Decision states that most DOE sites will continue to use offsite facilities for the treatment and disposal of major portions of their nonwastewater hazardous waste, and the Oak Ridge Reservation and SRS will continue to treat some of their own nonwastewater hazardous waste on site in existing facilities, where this is economically favorable. The high-level radioactive waste Record of Decision states that immobilized high-level radioactive waste will be stored at the site of generation. DOE decided in the Record of Decision for the management and disposal of low-level and mixed low-level radioactive waste to perform minimum treatment of low-level radioactive waste at all sites and continue, to the extent practicable, disposal of onsite low-level radioactive waste at INEEL, Los Alamos National

Laboratory, the Oak Ridge Reservation, and SRS. For the management and disposal of mixed low-level radioactive waste, DOE decided to treat this waste at the Hanford site, INEEL, the Oak Ridge Reservation, and SRS, with disposal at the Hanford site and the Nevada Test Site. More detailed information concerning DOE's alternatives for the future configuration of waste management facilities at INEEL is presented in the Waste Management Programmatic EIS and the transuranic, hazardous, high-level, and low-level and mixed low-level radioactive waste Records of Decision.

Table 3–14 Preferred INEEL Waste Management Alternatives From the Waste Management Programmatic EIS and Associated Records of Decision

Waste Type	*Preferred Action*
High-level radioactive	DOE prefers onsite storage of INEEL's immobilized high-level radioactive waste pending disposal in a geologic repository. [a]
Transuranic and mixed transuranic	DOE has decided that INEEL should prepare and store its transuranic waste on site pending disposal at the Waste Isolation Pilot Plant. [b]
Low-level radioactive	DOE has decided to treat INEEL low-level radioactive waste on site. INEEL has been selected as one of the regional disposal sites for low-level radioactive waste. [c]
Mixed	DOE has decided to treat INEEL mixed waste on site, including the possibility of treating mixed waste generated at other sites. INEEL was not selected as one of the regional disposal sites for mixed waste. [c]
Hazardous	DOE has decided to continue to use commercial facilities for treatment of INEEL nonwastewater hazardous waste and onsite facilities for treatment of wastewater hazardous waste. [d]

[a] From the Record of Decision for high-level radioactive waste (64 FR 46661).
[b] From the Record of Decision for transuranic waste (63 FR 3629).
[c] From the Record of Decision for low-level and mixed low-level radioactive waste (65 FR 10061).
[d] From the Record of Decision for hazardous waste (63 FR 41810).
Sources: DOE 1997a, 63 FR 3629, 63 FR 41810.

3.3 Savannah River Site

SRS is located on about 80,130 hectares (198,000 acres) in southwest South Carolina. The site is approximately 40 kilometers (25 miles) southeast of Augusta, Georgia, and 19 kilometers (12 miles) south of Aiken, South Carolina. First established in 1950, SRS has been involved in tritium operations and nuclear material production for more than 40 years. Today the site includes 16 major production, service, research, and development areas, not all of which are currently in operation. The site is owned by the Federal Government and is administered, managed, and controlled by DOE. It is bordered by the Savannah River to the southwest and includes portions of three South Carolina counties: Aiken, Allendale, and Barnwell.

There are more than 3,000 facilities at SRS, including 740 buildings with 511,000 square meters (5,500,000 square feet) of floor area. Major nuclear facilities at SRS include fuel and plutonium storage facilities; target fabrication facilities; nuclear material production reactors; chemical separation plants; a uranium fuel processing area; liquid high-level radioactive waste tank farms; a waste vitrification facility; and the Savannah River Technology Center. SRS processes nuclear materials into forms suitable for continued safe storage, use, or transportation to other DOE sites. Tritium recycling facilities at SRS empty tritium from expired reservoirs, purify it to eliminate the helium decay product, and fill replacement reservoirs with specification tritium for nuclear stockpile weapons. Filled reservoirs are delivered to Pantex for weapons assembly and directly to the U.S. Department of Defense to replace expired reservoirs. Historically, DOE has produced tritium at SRS, but has not produced any since 1988.

3.3.1 Land Resources

3.3.1.1 Land Use

Forest and agricultural land predominate in the areas bordering SRS (**Figure 3–5**). There are also significant open water and nonforested wetlands along the Savannah River Valley. Incorporated and industrial areas are the only other significant land uses. There is limited urban and residential development bordering SRS. The closest residences are to the west, north, and northeast within 61 meters (200 feet) of the site boundary. The three counties in which SRS is located, Aiken, Allendale, and Barnwell, have not zoned any of the site land.

Outdoor public recreation facilities are plentiful and varied in the SRS region. Included are the Sumter National Forest, 76 kilometers (47 miles) to the northwest; Santee National Wildlife Refuge, 80 kilometers (50 miles) to the east; and Strom Thurmond Reservoir, 69 kilometers (43 miles) to the northwest. There are also a number of state, county, and local parks in the region, most notably Redcliffe Plantation, Rivers Bridge, Barnwell and Aiken County State Parks in South Carolina, and Mistletoe State Park in Georgia. The Crackerneck Reserve, which occupies 4,047 hectares (10,000 acres) of SRS adjacent to the Savannah River, is open to public use (DOE 1999e).

Land use at SRS can be classified into three major categories: forest/undeveloped, water/wetlands, and developed facilities. Approximately 58,500 hectares (144,600 acres), or 73 percent of the site, is undeveloped. Wetlands, streams, and lakes account for 18,000 hectares (44,500 acres), or 22 percent of the site. Developed facilities, including production and support areas, roads, and utility corridors, encompass 4,000 hectares (9,900 acres), or 5 percent of SRS. Woodland areas are managed primarily for timber production. The U.S. Forest Service, under an interagency agreement with DOE, harvests about 730 hectares (1,800 acres) of timber from SRS each year. In 1972, DOE designated all of SRS as a National Environmental Research Park. The National Environmental Research Park is used by the national scientific community to study the impacts of human activities on the cypress swamp and hardwood forest ecosystems. DOE has set aside approximately 5,700 hectares (14,100 acres) of SRS exclusively for nondestructive environmental research.

Land use in F-Area is classified as heavy industrial. The many facilities located in this area historically have been associated with chemical and physical processes used to separate uranium, plutonium, and fission products (DOE 1996b). Of the many buildings situated in these areas, the F-Canyon is the dominant structure.

Land use in L-Area is classified as heavy industrial. Facilities located in the area historically have been associated with nuclear materials production for national defense. The L-Reactor was shut down in 1988 for safety upgrades and has not restarted (DOE 2000).

3.3.1.2 Visual Resources

The dominant viewshed in the vicinity of SRS consists mainly of agricultural land and forest, with some limited residential and industrial areas. The SRS landscape is characterized by wetlands and forested upland hills. DOE facilities are scattered throughout the site and are brightly lit at night. These facilities are generally not visible off site, as views are limited by rolling terrain, normally hazy atmospheric conditions, and heavy vegetation. The only areas visually impacted by the DOE facilities are those within the view corridors of State Highway 125 and SRS Road 1.

The developed areas and utility corridors (transmission lines and aboveground pipelines) of SRS are consistent with a Visual Resource Management Class IV rating, in which management activities dominate the view and are the focus of viewer attention (DOI 1986). The remainder of SRS generally ranges in Visual Resource Management rating from Class II to Class III. Management activities within these classes may be seen, but should not dominate the view.

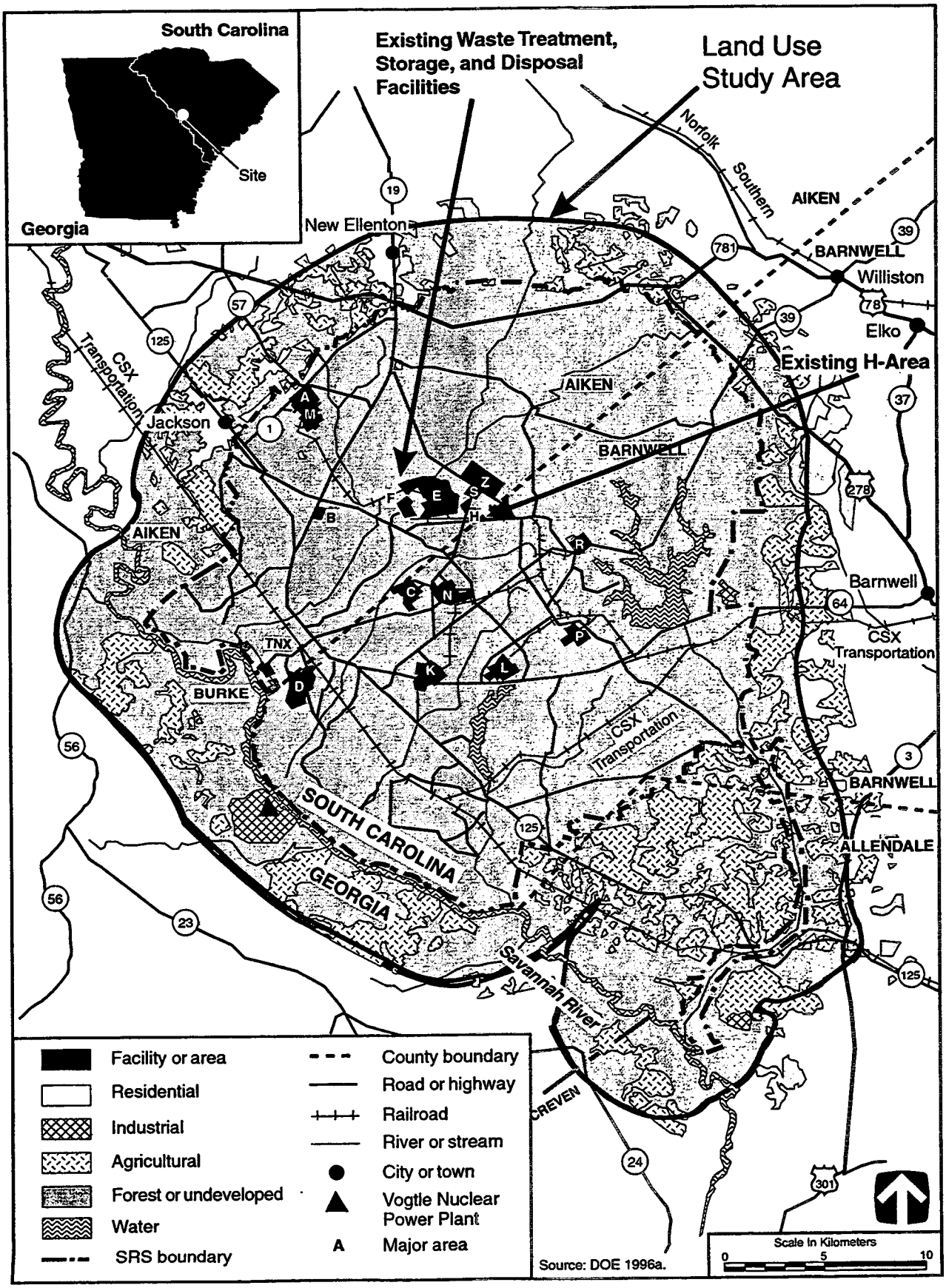

Figure 3–5 Generalized Land Use at SRS and Vicinity

Industrial facilities within F-Area and L-Area consist of large concrete structures, smaller administrative and support buildings, and parking lots. Structures generally range in height from 3 to 30 meters (10 to 100 feet). Facilities in these areas are brightly lit at night and are visible when approached via SRS access roads. However, neither area is visible from State Highway 125 or SRS Road 1 because of the distances involved and the presence of heavily wooded areas next to the roadways. Visual resource conditions in the F-Area and L-Area hold a Visual Resource Management Class IV rating.

3.3.2 Site Infrastructure

Site infrastructure includes those utilities and other resources required to support modification and continued operation of mission-related facilities identified under the various alternative actions. SRS comprises numerous research, processing, and administrative facilities. An extensive infrastructure system supports these facilities, as shown in **Table 3–15**.

Table 3–15 SRS-Wide Infrastructure Characteristics

Resource	*Current Usage*	*Site Capacity*
Transportation		
Roads (kilometers)	230	Not applicable
Railroads (kilometers)	103	Not applicable
Electricity		
Energy consumption (megawatt hours per year)	420,000	5,200,000
Peak load (megawatts)	70	330
Fuel		
Natural gas (cubic meters per year)	Not applicable	Not applicable
Oil (liters per year)	28,400,000	Not applicable [a]
Coal (tons per year)	210,000	Not applicable [a]
Water (liters per year)	1,780,000,000	3,870,000,000

[a] Low supplies can be replenished by truck or rail.
Source: DOE 1999h.

3.3.2.1 Transportation

SRS has an extensive network—230 kilometers (140 miles)—of roads to meet its onsite intrasite transportation requirements. The railroad infrastructure, which consists of 103 kilometers (64 miles) of track, provides deliveries of large volumes of coal and oversized structural components.

3.3.2.2 Electricity

The SRS electrical grid is a 115-kilovolt system in a ring arrangement that supplies power to operating areas, administrative areas, and independent and support function areas. That system includes about 160 kilometers (100 miles) of transmission lines. Power is supplied to the grid by three South Carolina Electric and Gas Company transmission lines. SRS is situated in and draws its power from the Virginia-Carolina Subregion, an electric power pool area that is a part of the Southeastern Electrical Reliability Council. Most of that power comes from offsite coal-fired and nuclear-powered generating plants.

Current site electricity consumption is about 420,000 megawatt hours per year. Site capacity is about 5.2 million megawatt hours per year. The peak load capacity is 330 megawatts; the peak load usage is 70 megawatts.

3.3.2.3 Fuel

Coal and oil are used at SRS primarily to power the steam plants. Steam generation facilities at SRS include coal-fired powerhouses at A-, D-, and H-Areas and two package steam boilers, which use No. 2 fuel oil, in K-Area. Coal is delivered by rail and is stored in coal piles in A-, D- and H-Areas. Oil is delivered by truck to K-Area. The A-Area powerhouse provides process and heating steam for the main administrative area at SRS. The D-Area powerhouse provides most of the steam for the SRS process area. Natural gas is not used at SRS.

3.3.2.4 Water

A new central domestic water system serves the majority of the site. The system includes: three wells and a 17-million-liter (4.5-million-gallon) per day water treatment plant in A-Area; two wells and an 8.3-million-liter (2.2-million-gallon) per day backup water treatment plant in B-Area; three elevated storage tanks; and a 43-kilometer (27-mile) piping loop. This central loop system has an estimated 1,680-liter (444-gallon) per minute excess capacity that could be increased by the installation of an additional elevated storage tank. Process water is provided to individual site areas.

3.3.2.5 Site Safety Services

The SRS fire department operates under a 12-hour rotational shift schedule, with three fire stations. Among the firefighters and officers are members of the SRS Hazardous Materials Response Team and the Rescue Team, who are responsible for rescues of all types. The fire department is supported by a fleet of 20 vehicles, including a specially prepared emergency response step van and trailer for hazardous materials response, and two boats for waterway spill response and control. Inspections are performed periodically according to National Fire Protection Codes and Standards.

3.3.3 Air Quality and Noise

3.3.3.1 Air Quality

Air pollution refers to any substance in the air that could harm human or animal populations, vegetation, or structures, or that unreasonably interferes with the comfortable enjoyment of life and property. Air pollutants are transported, dispersed, or concentrated by meteorological and topographical conditions. Air quality is affected by air pollutant emission characteristics, meteorology, and topography.

The SRS region has a temperate climate with short, mild winters and long, humid summers. Throughout the year, the climate frequently is affected by warm, moist maritime air masses. The average annual temperature at SRS is 17.3 °C (63.2 °F); temperatures vary from an average daily minimum of 0 °C (32 °F) in January to an average daily maximum of 33.2 °C (91.7 °F) in July. The average annual precipitation at SRS is 114 centimeters (45 inches). Precipitation is distributed fairly evenly throughout the year, with the highest in summer and the lowest in autumn. There is no predominant wind direction at SRS. The average annual wind speed at Augusta National Weather Service Station, the nearest National Weather Service Station, is 2.9 meters per second (6.5 miles per hour) (NOAA 1994).

Nonradiological Releases

SRS is near the center of the Augusta-Aiken Interstate Air Quality Control Region No. 53. None of the areas within SRS and its surrounding counties are designated as nonattainment areas with respect to the NAAQS for criteria air pollutants (40 CFR 50). Applicable NAAQS and state ambient air quality standards are presented in **Table 3–16.**

The primary emission sources of criteria air pollutants at SRS are the nine coal-burning boilers and four fuel oil-burning package boilers that produce steam and electricity, diesel engine-powered equipment, the Defense Waste Processing Facility, groundwater air strippers, the consolidated incineration facility, and various other process facilities. Other emissions and sources include fugitive particulates from coal piles and coal processing facilities, vehicles, controlled burning of forestry areas, and temporary emissions from various construction-related activities.

Table 3–16 presents the ambient air concentrations attributable to sources at SRS. These concentrations are based on dispersion modeling using emissions for the year 1998 (DOE 2000). Only those toxic and hazardous air pollutants that would be emitted for any of the alternatives analyzed in this EIS are presented. Concentrations shown in Table 3–16 are in compliance with applicable guidelines and regulations.

Table 3–16 Comparison of Modeled Ambient Air Concentrations at the SRS Boundary From SRS Sources With Most Stringent Applicable Standards or Guidelines

Pollutant	*Averaging Period*	*Most Stringent Standard or Guideline (micrograms per cubic meter)* [a]	*SRS Concentration (micrograms per cubic meter)*
Criteria Pollutants			
Carbon monoxide	8 hours	10,000 [b]	6,900
	1 hour	40,000 [b]	10,000
Nitrogen dioxide	Annual	100 [b]	2.6
Ozone	8 hours	157 [c,d]	(e)
PM_{10}	Annual	50 [b]	25
	24 hours (interim)	150 [b]	133
	24 hours (99^{th} percentile over 3 years)	150 [c]	(f)
$PM_{2.5}$	3-year annual	15 [c]	(f)
	24 hours (98^{th} percentile over 3 years)	65 [c]	(f)
Sulfur dioxide	Annual	80 [b]	34
	24 hours	365 [b]	350
	3 hours	1,300 [b]	200
State-Regulated Pollutants			
Gaseous fluoride	30 days	0.8 [g]	0.11
	7 days	1.6 [g]	0.6
	24 hours	2.9 [g]	1.2
	12 hours	3.7 [g]	2.4
Total suspended particulates	Annual	75 [g]	43.3
Hazardous/Toxic Compounds			
1,1,1-Trichloroethane	24 hours	9,550	22
Benzene	24 hours	150	3.9
Beryllium	24 hours	0.01	Less than 0.01
Biphenyl	24 hours	6	0.03
Ethanolamine	24 hours	200	Less than 0.01
Ethyl benzene	24 hours	4,350	0.12
Ethylene glycol	24 hours	650	0.08
Formaldehyde	24 hours	7.5	Less than 0.01
Glycol ethers	24 hours	Not applicable	Less than 0.01
Hexachloronaphthalene	24 hours	1	Less than 0.01
Hexane	24 hours	200	0.07
Manganese	24 hours	25	0.1
Mercury	24 hours	0.25	0.03
Methyl alcohol	24 hours	1,310	0.9
Methyl ethyl ketone	24 hours	14,750	0.99

Pollutant	Averaging Period	Most Stringent Standard or Guideline (micrograms per cubic meter) [a]	SRS Concentration (micrograms per cubic meter)
Methyl isobutyl ketone	24 hours	2,050	0.51
Methylene chloride	24 hours	515	1.8
Naphthalene	24 hours	1,250	0.01
Nitric acid	24 hours	125	6.7
Phenol	24 hours	190	0.03
Phosphorous	24 hours	0.5	Less than 0.01
Sodium hydroxide	24 hours	20	0.01
Toluene	24 hours	2,000	1.6
Trichloroethene	24 hours	6,750	1
Vinyl acetate	24 hours	176	0.02
Xylene	24 hours	4,350	3.8

PM_n = Particulate matter less than or equal to n microns in diameter.
[a] The more stringent of the Federal and state standards is presented if both exist for the averaging period.
[b] Federal and state standard.
[c] Standard currently under litigation, but will become enforceable during the life of the project.
[d] New NAAQS for ozone (8 hours limit of 0.08 parts per million [171 micro grams/cubic meter]) will become enforceable during the life of the project.
[e] Ambient concentrations of volatile organic compounds, which are precursors to ozone, can be used to provide a highly conservative bounding estimate for ozone but should not be used for explicit assessments of compliance with the ozone standard. Not all the volatile organic compounds emitted will result in the formation of ozone, and there is no method to directly correlate the two quantities. For purposes of estimating ozone concentrations from all SRS operations, no value for total volatile organic compounds is provided since the estimate would be overly conservative.
[f] No data is available with which to assess particulate matter concentrations.
[g] South Carolina state standard.
Sources: DOE 2000, Bickford et al. 1997, South Carolina R.62.5 (Standards 2 and 8), 40 CFR 50, 62 FR 38855, 62 FR 38652.

Data for 1995 from nearby South Carolina monitors at Jackson, Barnwell, and Beech Island (located 30 kilometers [18.6 miles] west of the site) indicate that the NAAQS for particulate matter, sulfur dioxide, and nitrogen dioxide are not exceeded in the area around SRS. Air pollutant measurements at these monitoring locations during 1995 showed: (1) for nitrogen dioxide, an annual average concentration of 9.4 micrograms per cubic meter; (2) for sulfur dioxide, concentrations of 99 micrograms per cubic meter for 3-hour averaging, 24 micrograms per cubic meter for 24-hour averaging, and 5 micrograms per cubic meter for the annual average; (3) for total suspended particulates, an annual average concentration of 37 micrograms per cubic meter; and (4) for PM_{10}, concentrations of 62 micrograms per cubic meter for 24-hour averaging and 19 micrograms per cubic meter for the annual average.

There are no Prevention of Significant Deterioration Class I areas within 100 kilometers (62 miles) of SRS. None of the facilities at SRS have been required to obtain a Prevention of Significant Deterioration permit (DOE 1996c). There are no Prevention of Significant Deterioration increment-consuming sources at SRS.

The meteorological conditions described for SRS are considered representative of F-Area and L-Area. The primary sources of nonradiological air emissions at F-Area and L-Area are diesel generators.

Radiological Releases

The primary sources of radiological air pollutants at SRS, as well as the localized releases associated with "separations" processing (e.g., F-Canyon), are presented in **Table 3–17** (Arnett and Mamatey 1998a). As shown in the table, tritium accounts for most of the total radioactivity released to the atmosphere from SRS operations. During 1997, about 58,000 curies of tritium (both in elemental and oxide forms) were released from SRS.

Table 3–17 Radiological Gaseous and Airborne Emissions at SRS in 1997 (Curies)

Effluent Type	*Radionuclide* [a]	*Reactors*	*Separations* [b]	*Other Facilities at SRS* [c]	*Total* [d]
Gases and vapors	Tritium (oxide)	5,230	33,400	506	39,136
	Tritium (elemental)	—	18,900	—	18,900
	Tritium (total)	5,230	52,300	506	58,036
	Carbon-14	—	0.031	—	0.031
	Krypton-85	—	9,620	—	9,620
	Iodine-129	—	0.0071	1.2×10^{-7}	0.0071
	Iodine-131	—	0.000029	0.00003	0.000059
	Iodine-133	—	—	0.00049	0.00049
Airborne particulates	Cobalt-57	—	2.1×10^{-7}	—	2.1×10^{-7}
	Cobalt-60	—	3.4×10^{-7}	9.1×10^{-7}	1.3×10^{-6}
	Strontium-89, 90	0.0018	0.00022	0.0003	0.0023
	Zirconium-95	—	—	0.000021	0.000021
	Ruthenium-106	—	—	0.07	0.07
	Antimony-125	—	—	5.9×10^{-7}	5.9×10^{-7}
	Cesium-134	—	1.4×10^{-6}	—	1.4×10^{-6}
	Cesium-137	0.00025	0.00042	0.0042	0.0049
	Cerium-144	—	4.2×10^{-6}	6.1×10^{-6}	0.00001
	Europium-154	—	1.5×10^{-7}	6.0×10^{-6}	6.6×10^{-6}
	Europium-155	—	4.9×10^{-6}	1.7×10^{-6}	6.6×10^{-6}
	Uranium-234	—	8.0×10^{-6}	0.000018	0.000027
	Uranium-235	—	6.3×10^{-7}	1.1×10^{-6}	1.8×10^{-6}
	Uranium-236	—	—	4.8×10^{-7}	4.8×10^{-7}
	Uranium-238	—	0.000019	0.000036	0.000056
	Neptunium-239	—	—	2.2×10^{-7}	2.2×10^{-7}
	Plutonium-238	—	0.000033	0.00036	0.00039
	Plutonium-239	0.00029	0.000051	0.000039	0.00038
	Plutonium-240	—	—	1.1×10^{-6}	1.1×10^{-6}
	Plutonium-241	—	—	0.000052	0.000052
	Americium-241	—	0.000014	8.8×10^{-7}	0.000015
	Americium-243	—	—	0.000018	0.000018
	Curium-244	—	0.000025	0.00013	0.00015
Total		**5,230**	**61,920**	**506.1**	**67,656.1**

[a] Release quantities greater than 10^{-7} curies are presented.
[b] Includes F- and H-Canyon, spent fuel storage at receiving basin, waste management, and tritium facilities.
[c] Other facilities include the Savannah River Technology Center, heavy water processing in D-Area, the reactor material area (M-Area), and other unmonitored diffuse and fugitive sources.
[d] Total might differ from sums due to rounding.
Source: Arnett and Mamatey 1998a.

3.3.3.2 Noise

Major noise sources at SRS are primarily in developed or active areas and include various industrial facilities, equipment, and machines. Most industrial facilities at SRS are far enough from the site boundary that noise levels from these sources at the boundary would not be measurable or would be barely distinguishable from background levels. Major noise emission sources outside of these active areas consist primarily of vehicles and rail operations.

An important contributor to noise levels is traffic to and from SRS along access highways through the nearby towns of New Ellenton, Jackson, and Aiken. Noise measurements recorded during 1989 and 1990 along State Route 125 in the town of Jackson, at a point about 15 meters (50 feet) from the roadway, indicate that the one-

hour equivalent sound level from traffic ranged from 48 to 72 decibels A-weighted. The estimated day-night average sound levels along this route were 66 decibels A-weighted for summer and 69 decibels A-weighted for winter. Similarly, noise measurements along State Route 19 in the town of New Ellenton at a point about 15 meters (50 feet) from the roadway indicate that the one-hour equivalent sound level from traffic ranged from 53 to 71 decibels A-weighted. The estimated day-night average sound levels along this route were 68 decibels A-weighted for summer and 67 decibels A-weighted for winter.

No distinguishing noise characteristics at F-Area and L-Area have been identified. These areas are 8 kilometers (5 miles) and 13 kilometers (8 miles) or more from the site boundary, respectively. Thus, contributions to noise levels at the site boundary from these areas are not measurable.

3.3.4 Water Resources

3.3.4.1 Surface Water

The largest river in the area of SRS is the Savannah River, which borders the site on the southwest. Six streams flow through SRS and discharge into the Savannah River: Upper Three Runs, Beaver Dam Creek, Fourmile Branch, Pen Branch, Steel Creek, and Lower Three Runs. Upper Three Runs has two tributaries, Tims Branch and Tinker Creek; Pen Branch has one tributary called Indian Grave Branch; and Steel Creek has one tributary called Meyers Branch **(Figure 3–6)**.

There are two manmade lakes at SRS: L-Lake, which discharges to Steel Creek, and Par Pond, which discharges to Lower Three Runs. Also, up to 350 to 400 Carolina bays (i.e., closed depressions capable of holding water) occur throughout the site. While none of these bays receive direct effluent discharge, some do receive stormwater runoff (DOE 1996c, DOE 2000, WSRC 1997b).

Water historically has been withdrawn from the Savannah River for use mainly as cooling water; some, however, has been used for domestic purposes. SRS currently withdraws about 140 billion liters (37 billion gallons) per year from the river. Most of this water is returned to the river through discharges to various tributaries (DOE 1996c).

The average flow of the Savannah River is 280 cubic meters (10,000 cubic feet) per second. Five large upstream reservoirs, Jocassee, Keowee, Hartwell, Richard B. Russell, and Strom Thurmond, regulate the flow in the Savannah River, thereby lessening the impacts of drought and flooding on users downstream (DOE 1995b).

Several communities in the area use the Savannah River as a source of domestic water. The nearest downstream domestic water intake is the Beaufort-Jasper Water Authority in South Carolina, which withdraws 0.23 cubic meters (8.1 cubic feet) per second to service about 51,000 people. Treated effluent is discharged to the Savannah River from upstream communities and from treatment facilities at SRS. The average annual volume of flow discharged by the sewage treatment facilities at SRS is about 700 million liters (185 million gallons) (DOE 1996c, Barghusen and Feit 1995). The F- and L-Area facilities are not located within a 100-year flood plain; there is no information available concerning 500-year flood plains (WSRC 1995).

A map showing the 100-year flood plain is presented as Figure 3–6. No federally designated Wild and Scenic Rivers occur within the site (Barghusen and Feit 1995).

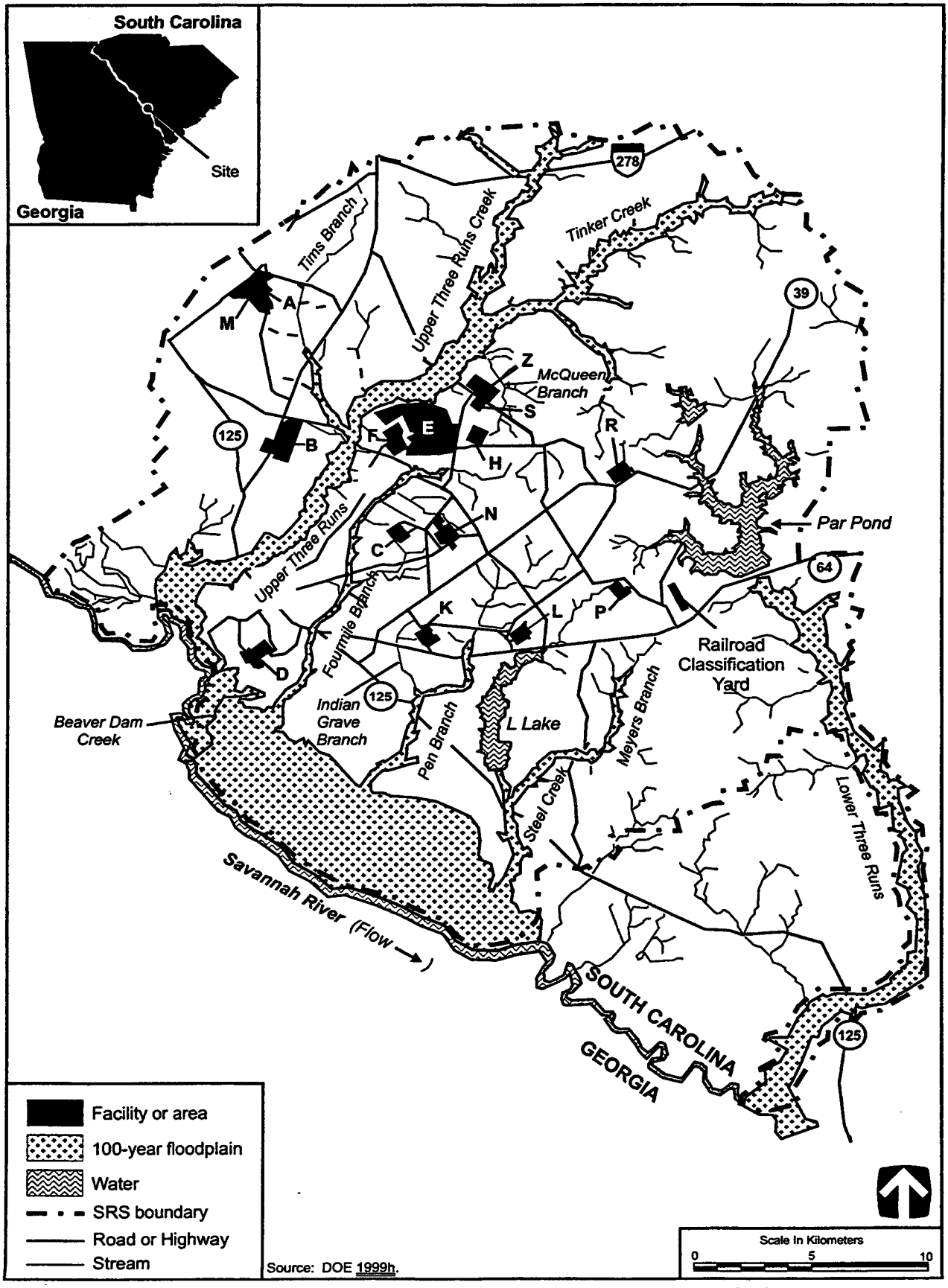

Figure 3–6 Locations of Water Bodies and Flood Plains at SRS

Nonradiological Releases

The Savannah River is classified as a freshwater source that is suitable for primary and secondary contact recreation; drinking, after appropriate treatment; fishing; balanced indigenous aquatic community development and propagation; and industrial and agricultural uses. A comparison of Savannah River water quality upstream (River Mile 160) and downstream (River Mile 120) of SRS showed no significant differences for nonradiological parameters. A comparison of 1997 data shows that the coliform data are within normal fluctuations for river water in this area and the overall cases in which standards were exceeded decreased in number from 1996 (Arnett and Mamatey 1998b). The data for the river's monitoring locations generally meet the freshwater standards set by the State of South Carolina; a comparison of the 1995 and earlier measurements for river samples showed no abnormal deviations.

Surface water rights for SRS are determined by the Doctrine of Riparian Rights, which allows owners of land adjacent to or under the water to use the water beneficially (DOE 1996c). SRS had five National Pollutant Discharge Elimination System (NPDES) permits in 1997, one (SC0000175) for industrial wastewater discharges, one (SCG250162) general permit for utility water discharge, two (SCR000000 and SCR100000) for general stormwater discharges, and one (ND0072125) for land application. Permit SC0000175 regulates 37 outfalls. The 1997 compliance rate for these outfalls was 99.9 percent. The 48 stormwater-only outfalls regulated by the stormwater permits are monitored as required. A pollution prevention plan has been developed to identify where the best available technology and best management practices must be used. For stormwater runoff from construction activities extending over 2 hectares (5 acres), a sediment reduction and erosion plan is required (Arnett and Mamatey 1996, Arnett and Mamatey 1997, Arnett and Mamatey 1998a).

The land around F-Area drains to Upper Three Runs and Fourmile Branch. Upper Three Runs is a large, cool blackwater stream that flows into the Savannah River. It drains about 54,390 hectares (134,400 acres) and has an average discharge of 9.3 cubic meters (330 cubic feet) per second near its mouth. The seven-day, 10-year low flow, which is the lowest flow over any seven days within any 10-year period, is 2.8 cubic meters (100 cubic feet) per second. The stream is about 40 kilometers (25 miles) long, yet only its lower reaches extend through SRS. It receives more water from underground sources than any other SRS stream and, therefore, has lower dissolved solids, hardness, and pH values. It is the only major stream on the site that has not received thermal discharges. It receives permitted discharges from several areas at SRS, including A-, B-, F-, H-, and S-Areas. Flow from the sanitary wastewater discharge averages less than 0.001 cubic meters (0.035 cubic feet) per second. A comparison with the seven-day, 10-year low flow of 2.8 cubic meters (100 cubic feet) per second in Upper Three Runs shows that the present discharges are very small (DOE 1994b, DOE 1995b).

Fourmile Branch is a blackwater stream affected by past operational practices at SRS. Its headwaters are near the center of the site, and it flows southwesterly before discharging into the Savannah River. The watershed is about 5,420 hectares (13,400 acres) and receives permitted effluent discharges from F-Area and H-Area. This stream received cooling water discharges from the C-Reactor while it was operating. Since those discharges ceased in 1985, the maximum recorded temperature in the stream has been 32 °C (90 °F), as opposed to ambient water temperatures that exceeded 60 °C (140 °F) when the reactor was operating. The average flow in the stream during the C-Reactor operation was 11.3 cubic meters (400 cubic feet) per second; since then, flows have averaged 1.8 cubic meters (64 cubic feet) per second. In its lower reaches, this stream widens and flows via braided channels through a delta. Downstream of this delta area, it reforms into one main channel, and most of the flow discharges into the Savannah River at River Mile 152, although a small portion flows west and enters Beaver Dam Creek. When the Savannah River floods, water from Fourmile Branch flows along the northern boundary of the flood plain and joins with other site streams to exit the swamp via Steel Creek instead of flowing directly into the Savannah River (DOE 1995b).

The land around L-Area drains to Steel Creek and Pen Branch. In its headwaters, Pen Branch is a largely undisturbed blackwater stream. Pen Branch and Indian Grave Branch drain an area of about 5,440 hectares (13,440 acres). Pen Branch flows southwesterly from its headwaters east of the K-Area to the Savannah River Swamp. At the swamp it flows parallel to the Savannah River for about 8 kilometers (5 miles) before it enters and mixes with Steel Creek. If the K-Reactor and its cooling tower were to operate, the flow in Indian Grave Branch would be reduced and a large part of its flow would be from cooling tower blowdown. This change would alter the water quality and temperature and flow regimes in Pen Branch. Currently, the Pen Branch system receives nonthermal effluent from K-Area and sanitary effluent from the Central Shops (N-Area). In water year 1991, the mean flow of Pen Branch at SC125 was 4.1 cubic meters (145 cubic feet) per second. Since the shutdown of the K-Reactor, the mean temperature of Pen Branch has been 22 °C (72 °F) and the flow at Road A-13.2 has averaged 0.55 cubic meters (19.3 cubic feet) per second (DOE 1995b; DOE 1997b).

The headwaters of Steel Creek originate near the P-Reactor. The creek flows approximately 3 kilometers (2 miles) before it enters the headwaters of L-Lake. L-Lake is 6.5 kilometers (4 miles) long with an area of about 420 hectares (1,040 acres). Flow from the outfall of L-Lake travels about 5 kilometers (3 miles) before entering Savannah River Swamp and then another 3 kilometers (1.9 miles) before entering the Savannah River. Myers Branch joins Steel Creek downstream of the L-Lake dam. The total area drained by the Steel Creek-Myers Branch system is about 9,070 hectares (22,400 acres). When the L-Reactor was operating, Steel Creek received cooling water from the L-Reactor, ash basin runoff, nonprocess cooling water, powerhouse wastewater, reactor process effluent, sanitary treatment plant effluent, and vehicle wash waters. During water year 1996, the mean flow rate of Steel Creek was 1.7 cubic meters (59.2 cubic feet) per second (DOE 2000).

Radiological Releases

Table 3–18 summarizes the radioactive liquid effluent released at SRS during 1997 (Arnett and Mamatey 1998a). As shown in the table, tritium accounts for most of the radioactivity discharged in SRS liquid effluent. In regard to actinides in nearby streams, trace amounts of uranium and plutonium were detected at a number of stream transport locations. Consequently, these small amounts were incorporated into the source term used for the calculation of the annual dose.

Table 3–18 Radiological Liquid Effluent at SRS in 1997 (Curies)

Radionuclide [a]	*Reactors*	*Separations* [b]	*Other Facilities at SRS* [c]	*Total* [d]
Tritium (oxide)	2,910	5,240	404	8,550
Strontium-89/90 [e]	0.065	0.14	0.0092	0.21
Iodine-129	—	0.078	—	0.078
Cesium-137	0.0029	0.045	—	0.048
Uranium-234	0.0045	0.023	0.00013	0.028
Uranium-235	0.000049	0.00072	3.6×10^{-6}	0.00078
Uranium-238	0.0038	0.026	0.00018	0.03
Plutonium-238	0.000042	0.00096	2.6×10^{-6}	0.001
Plutonium-239 [f]	0.011	0.034	5.6×10^{-6}	0.05
Americium-241	—	7.8×10^{-6}	2.1×10^{-6}	9.9×10^{-6}
Curium-244	—	2.9×10^{-6}	4.1×10^{-7}	3.3×10^{-6}

[a] Release quantities greater than 10^{-7} curies are presented.
[b] Representative of F- and H-Canyon operations, which include separations, waste management, and tritium facilities.
[c] Other facilities include the Savannah River Technology Center, heavy water processing in D-Area, the multipurpose pilot plant campus, and the reactor material area (M-Area).
[d] Totals might differ from sums due to rounding.
[e] Includes other unidentified beta emissions.
[f] Includes other unidentified alpha emissions.
Source: Arnett and Mamatey 1998a.

3.3.4.2 Groundwater

Aquifers are classified by Federal and state authorities according to use and quality. The Federal classifications include Class I, II, and III groundwater. Class I groundwater is either the sole source of drinking water or is ecologically vital. Class IIA and IIB are current or potential sources of drinking water (or other beneficial use), respectively. Class III is not considered a potential source of drinking water and is of limited beneficial use.

Although many different systems have been used to describe groundwater systems at SRS, for this EIS the system used in the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement* (DOE 1996c) has been adopted. The uppermost aquifer is referred to as the water table aquifer. It is supported by the leaky "Green Clay" aquitard, which confines the Congaree aquifer. Below the Congaree aquifer is the leaky Ellenton aquitard, which confines the Cretaceous aquifer, also known as the Tuscaloosa aquifer. In general, groundwater in the water table aquifer flows downward to the Congaree aquifer or discharges to nearby streams. Flow in the Congaree aquifer is downward to the Cretaceous aquifer or horizontal to stream discharge or the Savannah River, depending on the location within SRS.

Groundwater in the area is used extensively for domestic and industrial purposes. Most municipal and industrial water supplies in Aiken County are withdrawn from Cretaceous intermediate to deep aquifer units, while small domestic supplies are withdrawn from the water table aquifer. In Barnwell and Allendale counties, the Congaree aquifer supplies some municipal users. It is estimated that about 13 billion liters (3.4 billion gallons) per year are withdrawn from the aquifers within a 16-kilometer (10-mile) radius of the site, which is similar to the volume used by SRS (DOE 1996c). The Cretaceous aquifer is an important water resource for the SRS region. Aiken, South Carolina, for example, uses the Cretaceous aquifer for drinking water. The water is generally soft, slightly acidic, and low in dissolved and suspended solids (DOE 1995b).

Groundwater is the only source of domestic water at SRS (DOE 1995b). All groundwater at SRS is classified by the EPA as a Class II water source, and depth to groundwater ranges from near the surface to about 46 meters (150 feet) (DOE 1996c). SRS withdrawals of groundwater to support site operations range from 34,000 to 45,000 cubic meters (9 to 12 million gallons) per day (DOE 2000). There are no designated sole source aquifers in the area (Barghusen and Feit 1995).

Groundwater ranges in quality across the site. In some areas it meets drinking water quality standards, while in areas near some waste sites it does not. The Cretaceous aquifer is generally unaffected except for an area near A-Area, where trichlorothylene has been reported. Trichlorothylene also has been reported in the A- and M-Areas in the Congaree aquifer. Tritium has been reported in groundwater in the Separations Area. The water table aquifer is contaminated with solvents, metals, and low levels of radionuclides at several SRS sites and facilities. Groundwater eventually discharges into onsite streams or the Savannah River (DOE 1996c), but groundwater contamination has not been detected beyond SRS boundaries (DOE 1995b).

Groundwater rights in South Carolina are associated with the absolute ownership rule. Owners of land overlying a groundwater source are allowed to withdraw as much water as they desire; however, the state requires users who withdraw more than 379,000 liters (100,000 gallons) per day to report their withdrawals. SRS is required to report because its usage is above the reporting level (DOE 1996c).

Groundwater in the shallow, intermediate, and deep aquifers flows in different directions, depending on the depths of the streams that cut the aquifers. The shallow aquifer discharges to Upper Three Runs and Fourmile Branch. Shallow groundwater in the vicinity of F-Area flows toward Upper Three Runs, McQueen Branch, or Fourmile Branch. Groundwater in the intermediate and deep aquifers flows horizontally toward the Savannah River and southeast toward the coast (DOE 1994b).

Groundwater also moves vertically. In the shallow aquifer, it moves downward until its movement is obstructed by impermeable material. Operating under a different set of physical conditions, groundwater in the intermediate and deep aquifers flows mostly horizontally. Near F-Area, it moves upward because of higher water pressure below the confining unit between the upper and lower aquifers. This upward movement helps to protect the lower aquifers from contaminants found in the shallow aquifer. The depth to groundwater in F-Area varies from about 1 to 20 meters (3.3 to 66 feet) (DOE 1994b).

Groundwater quality in F-Area is not significantly different from that for the site as a whole. It is abundant, usually soft, slightly acidic, and low in dissolved solids. High dissolved iron concentrations occur in some aquifers. Where needed, groundwater is treated to raise the pH and remove iron (DOE 1994b).

Groundwater quality in the F-Area can exceed drinking water standards for several contaminants. Near the F-Area seepage basins and inactive process sewer line, radionuclide contamination is widespread. Most of these wells contain tritium above drinking water standards. Other wells exhibit gross alpha, gross beta, strontium-90, and iodine-129 above their standards. Other radionuclides found above proposed standards in several wells include americium-241; curium-243 and -244; radium-226 and -228; strontium-90; total alpha–emitting radium; and uranium-233, -234, -235, and -238. Cesium-137, curium-245 and -246, and plutonium-238 also were found (Arnett and Mamatey 1996).

Near the F-Area Tank Farm, tritium, mercury, nitrate-nitrite as nitrogen, cadmium, gross alpha, and lead were detected above drinking water standards in one or more wells. The pH exceeded the basic standard, and trichlorofluoromethane (freon-11), which has no drinking water standard, was present in elevated levels (Arnett and Mamatey 1996).

At the F-Area Sanitary Sludge Land Application Site, tritium, specific conductance, lead, and copper were found to exceed their drinking water standards in one or more wells. Groundwater near the F-Area Acid/Caustic Basin consistently exceeded drinking water standards for gross alpha. Total alpha-emitting radium, alkalinity, gross beta, nitrate as nitrogen, and pH were above their respective standards in one or more wells. The groundwater near the F-Area Coal Pile Runoff Containment Basin did not exceed any chemical or radiological standard during 1995 (Arnett and Mamatey 1996).

L-Area groundwater exceeds guidelines for tritium, other radionuclides, carbon disulfide, chlorinated and volatile organics, and metals. Groundwater beneath the L-Area Disassembly Basin has been contaminated with metals, chlorinated organics, and tritium (DOE 2000).

3.3.5 Geology and Soils

Coastal Plain sediments beneath SRS overlie a basement complex composed of Paleocene crystalline and Triassic sedimentary formations of the Dunbarton Basin. Small and discontinuous zones of calcareous sand (i.e., sand containing calcium carbonate [calcite]), which potentially is subject to dissolution by water, are beneath some parts of SRS. If dissolution occurs in these zones, potential underground subsidence resulting in settling of the ground surface could occur. No settling as a result of dissolution of these zones has been identified. No economically viable geologic resources have been identified at SRS.

In the immediate region of SRS, there are no known capable faults. A capable fault is one that has had movement at or near the ground surface at least once within the past 35,000 years or recurrent movement within the past 500,000 years. Several faults have been identified from subsurface mapping and seismic surveys within the Paleozoic and Triassic basement beneath SRS. These are shown in Figure 3.1-3 of the *Savannah River Site Spent Nuclear Fuel Management Final Environmental Impact Statement* (DOE 2000). The largest fault is the Pen Branch Fault. However, there is no evidence of movement within the last 38 million years along this fault.

Three earthquakes have occurred inside the SRS boundary between 1985 and 1997. The acceleration produced by these earthquakes did not activate seismic monitoring instruments in the reactor areas (these instruments have detection limits of 0.002 g). Existing information does not conclusively correlate these earthquakes with any of the known faults on the site (DOE 1999h). Historically, two large earthquakes have occurred within 160 kilometers (100 miles) of SRS. The Charleston earthquake of 1886 had an estimated Richter magnitude of 6.8, while the Union County, South Carolina, earthquake of 1913 had an estimated Richter magnitude of 6.0. The SRS area experienced an estimated peak horizontal acceleration of 0.10 g during the Charleston earthquake. An earthquake with a maximum horizontal acceleration of 0.2 g is estimated to have an annual probability of occurrence of 1 in 5,000 at SRS. An earthquake of this magnitude would not result in structural damage since this represents the design-basis earthquake (DOE 1995c).

There is no volcanic hazard at SRS. The area has not experienced volcanic activity within the last 230 million years. Future volcanism is not expected because SRS is along the passive continental margin of North America.

The soils at SRS are primarily sands and sandy loams. The somewhat excessively drained soils have a thick, sandy surface layer that extends to a depth of 2 meters (6.6 feet) or more in some areas. Soil units that meet the soil requirements for prime farmland soils exist on SRS. However, the U.S. Department of Agriculture's Natural Resources Conservation Service does not identify these as prime farmlands due to the nature of site use; that is, the lands are not available for the production of food or fiber. The soils at SRS are considered acceptable for standard construction techniques.

The soils of the F-Area and L-Area fall within the Fuquay-Blanton-Dothan Association. This association consists of nearly level to sloping, well-drained soils on broad upland ridges. Soils in this association have moderately thick, sandy surface and subsurface layers and a loamy subsoil (WSRC 1997b). Most soils within the F-Area and L-Area have been disturbed by site development activities.

3.3.6 Ecological Resources

Ecological resources include terrestrial resources, wetlands, aquatic resources, and threatened and endangered species. Material presented in this section, unless otherwise noted, is from the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement* (DOE 1996c).

3.3.6.1 Terrestrial Resources

Most of SRS has remained undeveloped since it was established in 1950. Only about 5 percent of the site is occupied by DOE facilities. Five major plant communities have been identified at SRS (**Figure 3-7**). Of these, the largest is the loblolly, longleaf, slash pine community, which covers approximately 65 percent of the site. This community type, as well as upland hardwood-scrub oak, occurs primarily in upland areas. Swamp forests and bottomland hardwood forests are found along the Savannah River and the numerous streams that traverse SRS. More than 1,300 taxa of vascular plants have been identified on the site.

Because of the variety of plant communities on the site, as well as the region's mild climate, SRS supports a diversity and abundance of wildlife, including 44 amphibian, 59 reptile, 255 bird, and 54 mammal species (DOE 1999h). Common species at SRS include the slimy salamander, eastern box turtle, Carolina chickadee, common crow, eastern cottontail, and gray fox. A number of game animals are found on SRS; however, except for the Crackerneck Reserve, only the whitetail deer and feral hog are hunted on site. Raptors, such as the Cooper's hawk and black vulture, and carnivores, such as the gray fox and raccoon, are ecologically important groups on SRS.

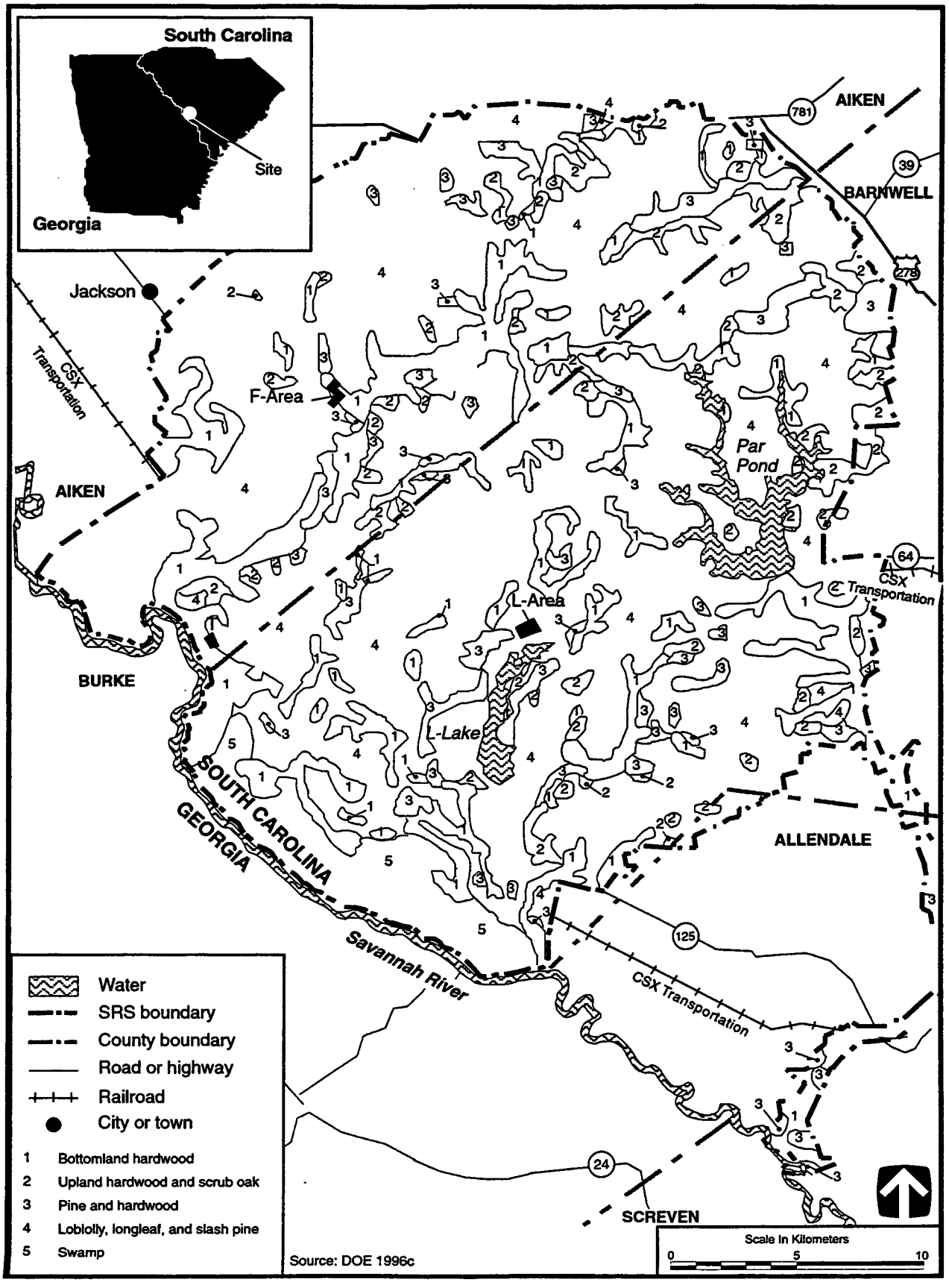

Figure 3–7 Distribution of Plant Communities at SRS

F-Area is an industrial area situated on an upland plateau between the drainage areas of Upper Three Runs and Fourmile Branch. It is surrounded primarily by evergreen forests with areas of grassland, scrub-shrub, and barren land also present. A roughly 6-hectare (15-acre) oak-hickory forest area designated as a National Environmental Research Park set aside is located northwest of the site. Bottomland hardwood forest areas are located along Upper Three Runs and Fourmile Branch. Buildings, paved parking lots, graveled construction areas, and laydown yards dominate this heavily industrialized area; little natural vegetation remains inside the fenced areas (DOE 1996b, DOE 1999h). A total of 41 animal species have been identified in and around F-Area, including 18 species of birds, 11 species of mammals, and 12 species of reptiles (WSRC 1997a).

L-Area is an industrial area largely surrounded by the loblolly, longleaf, and slash pine community, although an area of pine-hardwood community is located to the west. L-Area lies within the Steel Creek drainage just north of L-Lake (Figure 3–7). Plant communities found along Steel Creek include bottomland hardwood. While grassy areas occur within L-Area, it is largely disturbed with little vegetation. A total of 35 animal species have been identified in and around L-Area, including 15 species of birds, 8 species of mammals, and 12 species of reptiles (WSRC 1997a).

3.3.6.2 Wetlands

SRS contains approximately 19,800 hectares (49,000 acres) of wetlands, most of which are associated with flood plains, streams, and impoundments. Wetlands on the site may be divided into the following categories: bottomland hardwoods, cypress-tupelo, scrub-shrub, emergent, and open water. The most extensive wetland type on SRS is swamp forest associated with the Savannah River flood plain, which covers approximately 3,800 hectares (9,390 acres). Past releases of cooling water effluent into site streams and the Savannah River Swamp have resulted in shifts in plant community composition, including reduction in bottomland forests along streams and replacement of bald cypress by scrub-shrub and emergent vegetation in the swamp. As many as 350 to 400 Carolina bays, a type of wetland unique to the southeastern United States, also are found on SRS (DOE 1999c). These natural shallow depressions occur on interstream areas and range from lakes to shallow marshes, herbaceous bogs, shrub bogs, or swamp forests.

Wetlands in the vicinity of F-Area are associated primarily with Upper Three Runs and Fourmile Branch and their tributaries. These wetlands have been classified as bottomland hardwood. Below C-Area, Fourmile Branch was affected by cooling water discharged from the C-Reactor. These releases resulted in shifts in natural vegetation along the lower stream corridor and where it drains into the Savannah River Swamp. Since areas affected by shutdown of the reactor have revegetated, species composition is not the same as it was originally (WSRC 1997b).

Wetlands in the vicinity of L-Area are associated with Pen Branch, Steel Creek, and L-Lake. Prior to the establishment of SRS, wetlands associated with Pen Branch and Steel Creek were classified primarily as bottomland hardwood forest and swamp forest. Past releases of cooling water from the K-, L-, and P-Reactors resulted in shifts in plant community composition from bottomland forests along the stream corridors and cypress-tupelo in the Savannah River Swamp to scrub-shrub and emergent vegetation. Since shutdown of the reactors, some recovery of these areas has occurred; however, new growth has not always included the same species that were present in the original canopy. Wetlands associated with L-Lake include several shoreline zones, including a submersed and floating-leaf zone, emergent zone, and an upper emergent-shrub zone. Efforts have been made to revegetate both Ben Pranch and L-Lake (WSRC 1997b).

3.3.6.3 Aquatic Resources

Aquatic habitat on SRS includes manmade ponds, Carolina bays, reservoirs, and the Savannah River and its tributaries. There are more than 50 manmade impoundments located throughout the site that support populations of bass and sunfish. Fewer than 20 Carolina bays have permanent fish populations. Species

present in these bays include redfin pickerel, mud sunfish, lake chubsucker, and mosquitofish. Par Pond and L-Lake support similar fish populations, including largemouth bass, black crappie, and various species of pan fish. Sport fishing is permitted only within the Crackerneck Reserve. Commercial fishing is not allowed on SRS, although it does take place on the Savannah River. In the past, water intake structures for the C- and K-Reactors and the D-Area powerhouse caused annual estimated entrainment of approximately 10 percent of the fish eggs and larvae passing the intake canals during the spawning season. In addition, estimated impingement losses were approximately 7,600 fish per year.

Streams in the vicinity of F-Area include Upper Three Runs and Fourmile Branch and their tributaries. Fish species present in Upper Three Runs in the vicinity of F-Area include the dusky shiner, yellowfin shiner, redbreast sunfish, and bluegill. It is important as a spawning area for blueback herring and as a seasonal nursery habitat for American shad, striped bass, and other Savannah River species. Fish species present in Fourmile Branch near F-Area include the dusky shiner, creek chubsucker, yellow bullhead, and spotted sunfish. Studies of fish communities in Upper Three Runs and Fourmile Branch indicated that no measurable community-level impacts were associated with contaminants from the F-Area seepage basins (DOE 1996b, DOE 1999h).

Aquatic resources in the vicinity of L-Area are associated with Pen Branch, Steel Creek, and L-Lake. Pen Branch has been affected over the years by the operation and subsequent shutdown of the K-Reactor. During operations, fish populations in warmed portions of the stream were greatly reduced. With the end of reactor operations, a more diverse fish population has recolonized thermal portions of the stream. Steel Creek also has been affected by DOE operations, including the operation and subsequent shutdown of the L-Reactor, operation of the K-Reactor and the eventual diversion of its cooling waters to Par Pond, and the construction of L-Lake. L-Lake has undergone numerous changes in fish populations since it was first formed in 1985. These changes have been associated with colonization of the lake by fish originally in Steel Creek, as well as introduced fish and operation and eventual shutdown of the L-Reactor. Fish species that are common in the lake include largemouth bass, bluegill, redbreast sunfish, and threadfin shad (WSRC 1997b).

3.3.6.4 Threatened and Endangered Species

As shown in Table 3.7.6-1 in the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement* (DOE 1996a), 61 threatened, endangered, and other special status species listed by the Federal Government or the State of South Carolina may be found in the vicinity of SRS. Ten species are federally or state-listed as threatened or endangered (WSRC 1997b). No critical habitat for threatened or endangered species exists on SRS.

No federally listed threatened or endangered species are known to occur in F-Area, although several species may occur in the general vicinity. The American alligator (listed as threatened by virtue of its similarity in appearance to the endangered American crocodile), while fairly abundant on SRS, is uncommon in F-Area. The nearest active bald eagle nest is located along Pen Branch, 8 kilometers (5 miles) southeast of F-Area. Bald eagles are listed as threatened by the U.S. Fish and Wildlife Service and as endangered by South Carolina. Wood storks have been observed 14.5 kilometers (9 miles) from F-Area, near the Fourmile Branch delta. The closest colony of red-cockaded woodpeckers is 12 kilometers (7.5 miles) to the northeast, but suitable forage habitat exists near F-Area (WSRC 1997b). Both wood storks and red-cockaded woodpeckers are federally and state-listed as endangered. The smooth purple coneflower, the only endangered plant species found on SRS, has been found along Burma Road 4.8 kilometers (3 miles) southwest of F-Area. The state-listed rare Oconee azalea has been found on steep slopes adjacent to the Upper Three Runs flood plain just northwest of F-Area (DOE 1995b).

No federally listed threatened or endangered species are known to occur in L-Area, but several species may exist in the general vicinity. The American alligator has been observed in L-Lake and in Steel Creek below

L-Lake. Bald eagles have been observed in the L-Lake area; the nearest bald eagle nest is located on Pen Branch 3.2 kilometers (2 miles) southeast of L-Area. Wood storks have been observed in the Steel Creek delta, located about 9.8 kilometers (6 miles) south of L-Area. The closest colony of red-cockaded woodpeckers to L-Area is located about 8 kilometers (5 miles) to the east-southeast (WSRC 1997b). The nearest colony of the smooth purple coneflower to the site is located about 2.4 kilometers (1.5 miles) to the east near the junction of SRS Roads 9 and B. The Oconee azalea has been identified on the steep slopes adjacent to the Upper Three Runs flood plain about 12 kilometers (7.5 miles) northwest of L-Area (DOE 1995b). Consultation has been conducted with both the U.S. Fish and Wildlife Service and the state.

3.3.7 Cultural and Paleontological Resources

Cultural resources are human imprints on the landscape and are defined and protected by a series of Federal laws, regulations, and guidelines. Field studies conducted over the past two decades by the University of South Carolina's Institute of Archaeology and Anthropology have provided considerable information about the distribution and content of cultural resources at SRS. About 60 percent of SRS has been surveyed, and 858 archaeological (historic and prehistoric) sites have been identified. There are 67 sites considered potentially eligible for listing on the National Register; most of the sites have not been evaluated yet. No SRS nuclear production facilities have been nominated for the National Register, and there are no plans for nominations. Existing SRS facilities lack architectural integrity and do not contribute to the broad historic theme of the Manhattan Project and the production of World War II era nuclear materials.

Cultural resources at SRS are managed under the terms of a programmatic memorandum of agreement among the DOE Savannah River Operations Office, the South Carolina State Historic Preservation Officer, and the Advisory Council on Historic Preservation, dated August 24, 1990. Guidance on the management of cultural resources at SRS is included in the *Archaeological Resources Management Plan of the Savannah River Archaeological Research Program* (SRARP 1989).

3.3.7.1 Prehistoric Resources

Prehistoric resources are physical properties that remain from human activities that predate written records. Prehistoric resources at SRS consist of villages, base camps, limited-activity sites, quarries, and workshops. An extensive archaeological survey program begun at SRS in 1974 includes numerous field studies such as reconnaissance surveys, shovel test transects, and intensive site testing and excavation. There is evidence of more than 800 prehistoric sites, some of which may fall in the vicinity of the proposed facilities. Fewer than 8 percent of these sites have been evaluated for National Register eligibility.

Within F-Area, land areas have been disturbed over the past 46 years by activities associated with construction and operation of the existing facilities. Although no archaeological surveys have been conducted within the boundary of F-Area, no prehistoric cultural materials have been, or are expected to be, identified within this industrial area.

The potential for prehistoric sites in L-Area is limited. The area is in an archaeological site density zone that has the least potential for prehistoric sites of significance (DOE 2000).

3.3.7.2 Historic Resources

Historic resources consist of physical properties that postdate the existence of written records. Types of historic sites include farmsteads, tenant dwellings, mills, plantations and slave quarters, rice farm dikes, dams, cattle pens, ferry locations, towns, churches, schools, cemeteries, commercial building locations, and roads. About 400 historic sites or sites with historic components have been identified within SRS, and some of these may fall within the locations of the proposed facilities. To date, about 10 percent of the historic sites have been

evaluated for National Register eligibility. Most pre-SRS era historic structures were demolished during the initial establishment of SRS in 1950. Two SRS era buildings built in 1951 remain in use. From a Cold War perspective, SRS has been involved in tritium operations and other nuclear material production for more than 40 years; therefore, some existing facilities and engineering records may have significant historical and scientific content.

Within F-Area, land areas have been disturbed over the past 46 years by activities associated with the construction and operation of the existing facilities. Although no surveys have been conducted within the boundary of F-Area, no historic resources are expected to be identified, with the possible exception of surviving facilities and engineering records from the Cold War era.

The Savannah River Archaeological Research Program has not examined any areas in and immediately around Building 105-L. Archaeological resources in the footprint of the building are unlikely to have survived construction, although 1951 aerial photographs show that houses were present in L-Area before the development of SRS in the early 1950s (DOE 2000). Consultation has been conducted with the State Historic Preservation Office.

3.3.7.3 Native American Resources

Native American groups with traditional ties to the area include the Apalachee, Cherokee, Chickasaw, Creek, Shawnee, Westo, and Yuchi. At different times, each of these groups was encouraged by the English to settle in the area to provide protection from the French, Spanish, or other Native American groups. Main villages of both the Cherokee and Creek were located southwest and northwest of SRS, respectively, but both groups may have used the area for hunting and gathering activities. During the early 1800s, most of the remaining Native Americans residing in the region were relocated to the Oklahoma Territory.

Native American resources in the region include remains of villages or townsites, ceremonial lodges, burials, cemeteries, and natural areas containing traditional plants used in religious ceremonies. Literature reviews and consultations with Native American representatives have revealed concerns related to the American Indian Religious Freedom Act within the central Savannah River Valley, including some sensitive Native American resources and several plants traditionally used in ceremonies.

No onsite areas are subject to Native American Treaty Rights. However, five Native American groups, the Yuchi Tribal Organization, the National Council of Muskogee Creek, the Indian Peoples Muskogee Tribal Town Confederacy, the Pee Dee Indian Association, and the Ma Chis Lower Alabama Creek Indian Tribe, have expressed concern over sites and items of religious significance on SRS. DOE routinely notifies these organizations about major planned actions at SRS and asks them to comment on SRS documents prepared in accordance with NEPA.

In 1991, DOE conducted a survey of Native American concerns about religious rights in the central Savannah River Valley (DOE 1991). During this study, three Native American groups, the Yuchi Tribal Organization, the National Council of Muskogee Creek, and the Indian Peoples Muskogee Tribal Town Confederacy, expressed continuing interest in the SRS region with regard to the practice of their traditional religious beliefs. The Yuchi Tribal Organization and the National Council of Muskogee Creek have expressed concerns that several plant species (e.g., redroot, button snakeroot, and American ginseng) traditionally used in Tribal ceremonies could exist on SRS. Redroot and button snakeroot are known to occur on SRS, but are typically found in wet, sandy areas such as evergreen shrub bogs and savannas. Neither species is likely to be found in F-Area or L-Area because of past clearing associated with past development. In addition to those Native American Tribal organizations noted above, consultation has been conducted with the United Keetowah Band, the Pee Dee Indian Association, and the Ma Chris Lower Alabama Creek Indian Tribe.

3.3.7.4 Paleontological Resources

Paleontological resources are the physical remains, impressions, or traces of plants or animals from a former geologic age. Paleontological materials from the SRS area date largely from the Eocene Age (54 to 39 million years ago) and include fossil plants, numerous invertebrate fossils, giant oysters, other mollusks, and bryozoa. With the exception of the giant oysters, all other fossils are fairly widespread and common; therefore, the assemblages have low research potential or scientific value.

Paleontological resources have not been recorded in F-Area and their occurrence in L-Area is unlikely.

3.3.8 Socioeconomics

Statistics for employment and economy are presented for the regional economic area which encompasses 15 counties around SRS that are located in Georgia and South Carolina. Statistics for population and housing, community services, and local transportation are presented for the region of influence. The region of influence is a five-county area in which approximately 90 percent of all SRS employees reside (**Table 3–19**). In 1995, SRS employed 16,625 persons (6.5 percent of the 1996 regional economic area civilian labor force).

Table 3–19 Distribution of Employees by Place of Residence in the SRS Region of Influence, 1997

County	*Number of Employees*	*Total Site Employment (Percent)*
Aiken	8,966	53.9
Columbia	2,209	13.3
Richmond	2,204	13.3
Barnwell	1,112	6.7
Edgefield	242	1.5
Region of influence total	**14,733**	**88.6** [a]

[a] Total differs due to rounding.
Source: HNUS 1997.

3.3.8.1 Regional Economy Characteristics

Between 1990 and 1996, the civilian labor force in the regional economic area increased 3.6 percent to 257,101. In 1996, the unemployment rate in the regional economic area was 7.6 percent, which was greater than the unemployment rate of 6 percent for both Georgia and South Carolina.

In 1995, manufacturing represented the largest sector of employment in the regional economic area (25.6 percent). This was followed by government (20.9 percent) and service activities (19.9 percent). The total for these employment sectors in Georgia was 17.5 percent, 16.8 percent, and 23 percent, respectively. The total for these employment sectors in South Carolina was 23.3 percent, 17.3 percent, and 20.5 percent, respectively.

3.3.8.2 Population and Housing

In 1996, the region of influence estimated population totaled 453,778. Between 1990 to 1996, the region of influence population increased by 8.6 percent, compared with a 13 percent increase in Georgia's population and a 5.7 percent increase in South Carolina's population. Between 1980 and 1990, the number of housing units in the region of influence increased by 25.1 percent, compared with a 30.1 percent increase in Georgia and a 23.5 percent increase in South Carolina. The total number of housing units within the region of

influence for 1990 was 165,443 (DOE 1999h). In 1995, the total number of owner and renter housing units within the region of influence was 171,400 (DOE 1996c). The 1990 homeowner vacancy rate for the region of influence was 2.2 percent, compared with statewide rates of 2.5 percent for Georgia and 1.7 percent for South Carolina. The renter vacancy rate for the region of influence was 10 percent compared with the statewide rates of 12.2 percent for Georgia and 11.5 percent for South Carolina.

3.3.8.3 Community Services

Community services include public education and public safety. In 1997, school districts providing public education in the region of influence were operating at capacities of between 85 to 100 percent. Total student enrollment in the region of influence in 1997 was approximately 89,000, and the student-to-teacher ratio averaged 17 to 1. In 1990, the average student-to-teacher ratios were 10.8 to 1 for Georgia and 11.5 to 1 for South Carolina. In 1997, a total of 973 sworn police officers were serving the five-county region of influence. The average region of influence officer-to-population ratio was 2.1 officers per 1,000 persons. This compares with the 1990 state averages of 2 officers per 1,000 persons for Georgia and 1.8 officers per 1,000 persons for South Carolina.

3.3.8.4 Local Transportation

Vehicular access to SRS is provided by South Carolina State Routes 19, 64, and 125 (Figure 3–5). There is no public transportation to SRS. Rail service in the region of influence is provided by the Norfolk Southern Corporation and CSX Transportation. SRS is provided rail access via Robbins Station on the CSX Transportation line. Waterborne transportation is available via the Savannah River. SRS has no commercial docking facilities, but it has a boat ramp that has accepted large transport barge shipments. Columbia Metropolitan Airport in Columbia, South Carolina, and Bush Field in Augusta, Georgia, receive jet air passenger and cargo service from both national and local carriers.

3.3.9 Environmental Justice

Under Executive Order 12898, *Federal Actions to Address Environmental Justice in Minority Populations and Low-Income Populations*, Federal agencies are responsible for identifying and addressing the possibility of disproportionately high and adverse health or environmental effects of programs and policies on minority or low-income populations in potentially affected areas. Minority populations refer to all people of color, exclusive of white non-Hispanics. Low-income populations refer to households whose incomes are below the Federal poverty thresholds. In the case of SRS, the potentially affected area includes parts of Georgia and South Carolina.

Data obtained during the 1990 census show that the percentage of minorities for the contiguous United States was 24.1, and the percentages for the States of Georgia and South Carolina were 29.8 and 31.4, respectively. The same census data also show that, of the total population of the contiguous United States, 13.1 percent reported incomes below the poverty threshold, and Georgia and South Carolina reported 14.7 and 15.4 percent, respectively.

The potentially affected area surrounding F-Area is defined by a circle with an 80-kilometer (50-mile) radius centered at Building 221–F (latitude 33°17'11" N, longitude 81°40'38" W). The total population residing within that area in 1990 was 615,734. The proportion of the population around this building that was considered minority was 37.9 percent. At the time of the 1990 census, Blacks were the largest minority group within the potentially affected area, constituting 35.7 percent of the total population. Hispanics constituted about 1 percent, and Asians about 1 percent. Native Americans constituted about 0.2 percent of the population (DOC 1992).

A breakdown of incomes in the potentially affected area is also available from the 1990 census data (DOC 1992). At that time, the poverty threshold was $9,981 for a family of three with one related child under 18 years of age. A total of 107,479 persons (18 percent of the total population) residing within the potentially affected area around F-Area reported incomes below the poverty threshold.

The potentially affected area surrounding L-Area is defined by a circle with a radius equal to 80 kilometers (50 miles) centered at Building 105-L (latitude 33°12'38.5" N and longitude 81°37'26.5" W). The total population residing within the potentially affected area in 1990 was 606,819 persons. Approximately 39.1 percent of the population in 1990 was composed of individuals who identified themselves as having racial or ethnic origins that are used by the Council on Environmental Quality to define minority populations (CEQ 1997). At the time of the 1990 census, Blacks were the largest minority group within the potentially affected area, constituting approximately 36.8 percent of the total population. Less that 3 percent of the total population in the potentially affected area designated themselves as Asian, Native American, or Hispanic (DOC 1992).

Within the potentially affected area in 1990, 107,468 persons (nearly 21 percent of the total population) reported incomes that were less than the threshold for poverty.

3.3.10 Existing Human Health Risk

Public and occupational health and safety issues include the determination of potentially adverse effects on human health that result from acute and chronic exposures to ionizing radiation and hazardous chemicals.

3.3.10.1 Radiation Exposure and Risk

Major sources and levels of background radiation exposure to individuals in the vicinity of SRS are shown in **Table 3–20**. Annual background radiation doses to individuals are expected to remain constant over time. The total dose to the population, in terms of person-rem, changes as the population size changes. Background radiation doses are unrelated to SRS operations.

Table 3–20 Sources of Radiation Exposure to Individuals in the SRS Vicinity Unrelated to SRS Operations

Source	*Effective Dose Equivalent (millirem per year)*
Natural Background Radiation [a]	
Cosmic radiation	27
External terrestrial radiation	28
Internal terrestrial/cosmogenic radiation	40
Radon in homes (inhaled)	200 [b]
Other Background Radiation [c]	
Diagnostic x-rays and nuclear medicine	53
Weapons test fallout	less than 1
Air travel	1
Consumer and industrial products	10
Total	360

[a] Arnett and Mamatey 1998a.
[b] An average for the United States.
[c] NCRP 1987.

Releases of radionuclides to the environment from SRS operations provide another source of radiation exposure to individuals in the vicinity of SRS. Types and quantities of radionuclides released from SRS operations in 1997 are listed in the *Savannah River Site Environmental Report for 1997* (Arnett and Mamatey 1998a). The doses to the public resulting from these releases are presented in **Table 3–21**. These doses fall within radiological limits per DOE Order 5400.5, *Radiation Protection of the Public and Environment*, and are much lower than those of background radiation.

Table 3–21 Radiation Doses to the Public From Normal SRS Operations in 1997 (Total Effective Dose Equivalent)

Members of the Public	*Atmospheric Releases*		*Liquid Releases*		*Total*	
	Standard [a]	*Actual*	*Standard* [a]	*Actual* [b]	*Standard* [a]	*Actual*
Maximally exposed offsite individual (millirem)	10	0.050	4	0.13	100	0.18
Population within 80 kilometers (50 miles) (person-rem) [c]	None	5.5 [d]	None	2.4	100	7.9
Average individual within 80 kilometers (50 miles) (millirem) [c]	None	0.0089	None	0.0035	None	0.013

[a] The standards for individuals are given in DOE Order 5400.5. As discussed in that Order, the 10-millirem-per-year limit from airborne emissions is required by the Clean Air Act (40 CFR 61), and the 4-millirem-per-year limit is required by the Safe Drinking Water Act. For this EIS, the 4-millirem-per-year value is assumed conservatively to be the limit for the sum of doses from all liquid pathways. The total dose of 100 millirem per year is the limit from all pathways combined. The 100 person-rem value for the population is given in proposed 10 CFR 834, as published in 58 FR 16268. If the potential total dose exceeds the 100 person-rem value, the contractor operating the facility is required to notify DOE.

[b] Conservatively includes all water pathways, not just the drinking water pathway. The population dose includes contributions to Savannah River users downstream of SRS to the Atlantic Ocean.

[c] About 620,100 in 1997. For liquid releases, an additional 70,000 water users in Port Wentworth, Georgia, and Beaufort, South Carolina (about 160 kilometers [98 miles] downstream), are included in the assessment.

[d] This corresponds to the value calculated for Clean Air Act (40 CFR 61) compliance and is consistent with the assumptions used in dose calculations presented in this EIS.

[e] Obtained by dividing the population dose by the number of people living within 80 kilometers (50 miles) of the site for atmospheric releases; for liquid releases, the number of people includes water users who live more than 80 kilometers (50 miles) downstream of the site.

Source: Arnett and Mamatey 1998a.

Using a risk estimator of 500 latent cancer deaths per 1 million person-rem to the public (see Appendix E), the fatal cancer risk to the maximally exposed offsite individual resulting from radiological releases from SRS operations in 1997 is estimated to be 9.0×10^{-8}. That is, the estimated probability of this person dying of cancer at some point in the future from radiation exposure associated with one year of SRS operations is less than 1 in 10 million (it takes several to many years from the time of radiation exposure for a cancer to manifest itself).

According to the same risk estimator, 0.004 excess latent fatal cancers are projected in the population living within 80 kilometers (50 miles) of SRS from normal operations in 1997. To place this number in perspective, it may be compared with the number of fatal cancers expected in the same population from all causes. The 1995 mortality rate associated with cancer for the entire U.S. population was 0.2 percent per year. Based on this mortality rate, the number of latent fatal cancers expected during 1997 from all causes in the population living within 80 kilometers (50 miles) of SRS was 1,240. This expected number of fatal cancers is much higher than the 0.004 latent fatal cancers estimated from SRS operations in 1997.

SRS workers receive the same dose as the general public from background radiation, but they also receive an additional dose from working in facilities with nuclear materials. The average dose to the individual worker

and the cumulative dose to all workers at SRS from operations in 1997 are presented in **Table 3–22.** These doses fall within the radiological regulatory limits of 10 CFR 835. According to a risk estimator of 400 latent fatal cancers per 1 million person-rem among workers (Appendix E), the number of projected fatal cancers among SRS workers from normal operations in 1997 is 0.066. The risk estimator for workers is lower than the estimator for the public because of the absence from the work force of the more radiosensitive infant and child age groups.

Table 3–22 Radiation Doses to Workers From Normal SRS Operations in 1997 (Total Effective Dose Equivalent)

Occupational Personnel	*Onsite Releases and Direct Radiation*	
	Standard [a]	*Actual*
Average radiation worker (millirem)	None [b]	50
Total workers (person-rem)[c]	None	165

[a] The radiological limit for an individual worker is 5,000 millirem per year. However, DOE's goal is to maintain radiological exposure as low as is reasonably achievable. Therefore, DOE has established an administrative control level of 2,000 millirem per year (DOE Order N 441.1); the site must make reasonable attempts to maintain individual worker doses below this level.
[b] No standard is specified for an "average radiation worker;" however, the maximum dose that this worker may receive is limited to that given in footnote "a."
[c] In 1997, 3,327 workers with measurable doses.
Sources: DOE 1995a, DOE 1998d.

A more detailed presentation of the radiation environment, including background exposures and radiological releases and doses, is presented in the *Savannah River Site Environmental Report for 1997* (Arnett and Mamatey 1998a). The concentrations of radioactivity in various environmental media (including air, water, and soil) in the site region (on and off the site) also are presented in that report.

External radiation doses and concentrations of gross alpha, plutonium, and americium in air have been measured in F-Area. Onsite doses are measured for comparison against natural background levels, which are measured at offsite locations; the numerical difference in these measurements may be directly attributable to radiological sources that are located in the vicinity of the onsite measurement location(s). In 1997, the annual dose in F-Area was 105 millirem. This is about 20 millirem higher than the average dose measured at offsite locations. In the same year, the concentration of gross alpha was about 0.0011 picocuries per cubic meter in F-Area, compared with the approximately 0.00099 picocuries per cubic meter measured at the offsite control location. No plutonium-239 was detected in F-Area. Offsite controls also did not detect any plutonium-239 in the air in 1997 (Arnett and Mamatey 1998b).

External radiation doses have been measured in L-Area. In 1997, the annual dose in L-Area was 80 millirem (Arnett and Mamatey 1998b).

3.3.10.2 Chemical Environment

Table 3-16 identifies the hazardous (i.e., carcinogenic and toxic/noncarcinogenic) chemicals that are emitted to the air at SRS. The list includes only those chemicals that have ambient air quality standards and would be emitted under the alternatives analyzed at SRS. This list includes 24 chemicals, including benzene, ethyl benzene, formaldehyde, hexane, manganese, methyl-ethyl-ketone, methylene chloride, naphthalene, toluene, and vinyl acetate (see Table 3-17 for the complete list).

Health impacts on the public may occur by inhaling air containing hazardous chemicals, ingesting contaminated drinking water or food, and direct exposure (skin contact). The primary health impacts from

exposure to hazardous chemicals are from inhalation. Two major health effects are observed from the listed chemicals, the carcinogenic effect and the noncarcinogenic effect. These are presented below.

Carcinogenic Effects: These effects are estimated as the incremental probability of an individual developing cancer over a lifetime as a result of exposure to the potential carcinogen. This could be an incremental or excess individual lifetime cancer risk.

Noncarcinogenic Effects: These effects are determined by the ratio between the calculated or measured concentration of the chemical in the air and the reference concentration or dose. This ratio is known as the hazard quotient. Hazard quotients for noncarcinogens are summed to obtain the hazard index. If the hazard quotient is less than 1, then no adverse health effects are expected.

For some chemicals where the weight of evidence is weak and carcinogenecity is not well established, the impacts of both cancer and noncancer effects were determined. **Table 3-23** summarizes the baseline hazardous chemical impacts to the public. This table lists only those chemicals for which reference concentrations for cancer or toxicity are available from the Integrated Risk Information System. The baseline concentrations are estimates of the highest existing concentrations and represent the highest concentrations to which individuals from the public could be exposed under normal operations (excluding accident conditions). These concentrations are in compliance with applicable guidelines. Additional information on estimating the health impacts of hazardous chemicals is presented in Appendix E, Section E.5.

The exposure of workers to hazardous chemicals varies among facilities and the operational activities, and the available information is insufficient for a meaningful estimate of impacts. Workers are protected by adherence to the OSHA and EPA standards that regulate workplace atmospheric and drinking water concentrations of potentially hazardous chemicals. Monitoring the frequency and amount of chemicals released in operational processes ensures that these standards are not exceeded. Further, DOE requires that the environment in the workplace be as free as possible from recognized hazards that cause, or are likely to cause, illness or physical harm. Therefore, workplace conditions at SRS are substantially better than required by standards.

Table 3–23 Hazardous Chemical Impacts to the Public From Existing Activities at SRS

Chemical	*Annual Concentration (milligrams per cubic meter)*	*Reference Concentration Inhalation (milligrams per cubic meter)*	*Unit Cancer Risk (risk per milligram per cubic meter)*	*Hazard Quotient*	*Cancer Risk*
Benzene	0.0039	None	0.0078	None	0.00003
Ethyl benzene	0.000015	1	None	0.000015	None
Formaldehyde	1.3×10^{-6}	None	0.013	None	1.6×10^{-8}
Hexane	8.75×10^{-6}	0.2	None	0.000044	None
Manganese	0.000013	0.00005	None	0.25	None
Methyl-ethyl-ketone	0.00012	1	None	0.00012	None
Methylene chloride	0.00023	None	0.00047	None	1.1×10^{-7}
Naphthalene	1.3×10^{-6}	0.003	None	0.00042	None
Toluene	0.0002	0.4	None	0.0005	None
Vinyl acetate	2.5×10^{-6}	0.2	None	0.000013	None
Hazard Index				0.25	Not applicable

Sources: EPA 1999, Bickford et al. 1997.

3.3.10.3 Health Effects Studies

One epidemiological study on the general population in communities surrounding SRS has been conducted and published. No evidence of excess cancer mortality, congenital anomalies, birth defects, early infancy deaths, strokes, or cardiovascular deaths was reported. The epidemiological literature on the facility reflects an excess of leukemia deaths among hourly workers; no other health effects for workers are reported. For a more detailed description of the studies reviewed and their findings, and for a discussion of the epidemiologic surveillance program implemented by DOE to monitor the health of current SRS workers, refer to Appendix M.4.7 of the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement* (DOE 1996c).

3.3.10.4 Accident History

Between 1974 and 1988, there were 13 inadvertent tritium releases from the SRS tritium facilities. These releases were attributed to aging equipment in the tritium processing facility and are one of the reasons for the construction of the Replacement Tritium Facility at SRS. A detailed description and study of these incidents and their consequences for the offsite population have been documented by SRS. The most significant were in 1981, 1984, and 1985, when respectively 32,934, 43,800, and 19,403 curies of tritiated water vapor were released. From 1989 through 1992, there were 20 inadvertent releases, all with little or no offsite dose consequences. The largest of the recent releases occurred in 1992 when 12,000 curies of tritium were released.

3.3.10.5 Emergency Preparedness

Each DOE site has established an emergency management program that would be activated in the event of an accident. This program has been developed and maintained to ensure adequate response to most accident conditions and to provide response efforts for accidents not specifically considered. The emergency management program includes emergency planning, preparedness, and response.

The Emergency Preparedness Facility at SRS provides overall direction and control for onsite responses to emergencies and coordinates with Federal, state, and local agencies and officials on the technical aspects of the emergency. Emergency plans have been prepared for specific areas at SRS. Participating government agencies whose plans are interrelated with the SRS emergency plan for action include the States of South Carolina and Georgia, the City of Aiken, and the various counties in the general region of the site. Emergency response support, including firefighting and medical assistance, would be provided by these jurisdictions.

In addition, DOE has specified actions to be taken at all DOE sites to implement lessons learned from the emergency response to an accidental explosion at Hanford in May 1997.

3.3.11 Waste Management

Waste management includes minimization, characterization, treatment, storage, transportation, and disposal of waste generated from ongoing DOE activities. The waste is managed according to appropriate treatment, storage, and disposal technologies, and in compliance with all applicable Federal and state statutes and DOE Orders.

3.3.11.1 Waste Inventories and Activities

SRS manages the following types of waste: high-level radioactive, transuranic, mixed transuranic, low-level radioactive, mixed, hazardous, and nonhazardous. Waste generation rates and the inventory of stored waste from activities at SRS are provided in **Table 3–24**. **Table 3–25** summarizes the SRS waste management capabilities. More detailed descriptions of the waste management system capabilities at SRS are included in

Table 3–24 Waste Generation Rates and Inventories at SRS

Waste Type	Generation Rate (cubic meters per year)	Inventory (cubic meters)
High-Level Radioactive	1,561	131,000
Transuranic [a]		
Contact-handled	427	6,977
Remotely handled	4	0
Low-Level Radioactive	10,043	1,616
Mixed		
RCRA	1,135	6,940
Toxic Substances Control Act	0	110
Hazardous	74	1,416
Nonhazardous		
Liquid	416,100	Not applicable [b]
Solid	6,670	Not applicable [b]

[a] Includes mixed transuranic waste.
[b] Generally, nonhazardous waste is not held in long-term storage.
Sources: DOE 1999h, except high-level radioactive waste generations rates (DOE 1996c) and high-level radioactive waste inventory (DOE 1997a).

Table 3–25 Waste Management Capabilities at SRS

Facility Name/Description	Capacity	Status	Applicable Waste Type						
			HLW	TRU	Mixed TRU	LLW	Mixed	Haz	Non-Haz
Treatment Facility (cubic meters per year)									
Savannah River Technology Center Ion Exchangers, Evaporators	53,700	On-line	X						
Transuranic Waste Characterization/ Certification Facility	1,720	Planned for 2007		X	X				
Consolidated Incineration Facility and Ashcrete Stabilization Facility	4,630 liquid 17,830 solid	On-line				X	X	X	
F- and H-Area Effluent Treatment Facility	1,930,000	On-line				X	X		
M-, L-, and H-Area Compactors	3,983	On-line				X			
Non-Alpha Vitrification Facility	3,090	Planned				X	X	X	
M-Area Liquid Effluent Treatment Facility	999,000	On-line					X		
M-Area Vendor Treatment Facility	2,470	Planned					X		
Savannah River Technology Center Ion Exchange Treatment Probe	11,200	On-line					X		
Area Supercompactor	5,700	Planned				X			
Z-Area Saltstone Facility	28,400	On-line					X		
Central Sanitary Wastewater Treatment Facility	1,030,000	On-line							X

Facility Name/Description	Capacity	Status	Applicable Waste Type						
			HLW	TRU	Mixed TRU	LLW	Mixed	Haz	Non-Haz
Storage Facility (cubic meters)									
Transuranic Storage Pads	34,400	On-line		X	X				
Defense Waste Processing Facility Organic Waste Storage Tank	568	On-line					X		
Liquid Waste Solvent Tanks	454	Planned					X		
M-Area Process Waste Interim Treatment/Storage Facility	8,300	On-line					X		
Mixed Waste Storage Facilities (645-2N, -295, -43E)	1,905	On-line					X		
Savannah River Technology Center Mixed Waste Storage Tanks	198	On-line					X		
Long-Lived Waste Storage Building	1,064	Planned				X			
Solid Waste Storage Pads	2,657	On-line					X	X	
Buildings 316-M, 710-B, 645-N, and 645-4N	2,515	On-line					X	X	
M-Area Storage Pad	2,160	On-line					X		
F- and H-Area Tank Farm	133,000	On-line	X						
Defense Waste Processing Facility	2,286 canisters	On-line	X						
Disposal Facility (cubic meters)									
Intermediate-Level Radioactive Waste Vaults	3,665	On-line				X			
Low-Activity Waste Vaults	30,500	On-line				X			
Low-Level Radioactive Waste Disposal Facility Slit Trenches	26,000	Planned				X			
Z-Area Saltstone Vaults	1,110,000	On-line				X			

DWPF = Defense Waste Processing Facility, Haz = hazardous, HLW = high-level radioactive waste, LLW = low-level radioactive waste, TRU = transuranic
Sources: DOE 1999h, except high-level radioactive waste (DOE 1996c).

the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement* (DOE 1996c) and the *Savannah River Site Waste Management Final Environmental Impact Statement* (DOE 1995b).

The EPA placed SRS on the National Priorities List in December 1989. In accordance with CERCLA, DOE entered into a Federal Facilities Compliance Agreement with the EPA and the State of South Carolina to coordinate cleanup activities at SRS under one comprehensive strategy. As stated in the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement* (DOE 1996c), this Agreement combines the RCRA Facility Investigation Program Plan with a CERCLA cleanup program titled the *RCRA Facility Investigation/Remedial Investigation Program Plan*. More information on regulatory requirements for waste disposal is provided in Chapter 5.

3.3.11.2 High-Level Radioactive Waste

Liquid high-level radioactive waste at SRS is made up of many waste streams generated during the recovery and purification of transuranic waste products and unburned fissile material from spent reactor fuel elements. This waste is separated by waste form, radionuclide, and heat content before their transfer to underground

storage tanks in the F- and H-Area Tank Farms. Processes routinely used to treat liquid high-level radioactive waste are separation, evaporation, and ion exchange. Evaporation produces a cesium-contaminated condensate. Cesium is removed from the condensate, resulting in a low-level radioactive waste stream that is treated in the Effluent Treatment Facility. The remaining high-level radioactive waste stream salts are precipitated; some can be decontaminated. The decontaminated salt solution is sent with residues from the Effluent Treatment Facility to the Defense Waste Processing Z-Area Saltstone Facility, where it is mixed with a blend of cement, fly ash, and blast furnace slag to form grout. The grout is pumped into disposal vaults where it hardens for permanent disposal as solid low-level radioactive waste. The remaining high-level radioactive salt and sludge are immobilized permanently as a glass solid cast in stainless steel containers at the Defense Waste Processing Facility Vitrification Plant. The stainless steel containers are decontaminated to U.S. Department of Transportation standards, welded closed, and temporarily stored on site for eventual transport to and disposal in a repository. Future high-level radioactive waste generation could result from the processing and stabilization of spent nuclear fuel for long-term storage as a result of the Record of Decision (60 FR 28680) on the Programmatic Spent Nuclear Fuel EIS (DOE 1995a), and from remediation or materials recovery activities performed in the F- and H-Canyons.

3.3.11.3 Transuranic Waste

Transuranic waste generated between 1974 and 1986 is stored on five concrete pads and one asphalt pad that have been covered with approximately 1.2 meters (4 feet) of soil. Transuranic waste generated since 1986 is stored on 13 concrete pads that are not covered with soil. The transuranic waste storage pads are in the Low-Level Radioactive Waste Disposal Facility (DOE 1999h).

A planned Transuranic Waste Characterization and Certification Facility would provide extensive containerized waste certification capabilities. The facility is needed to prepare transuranic waste for treatment and to certify transuranic waste for disposal at the Waste Isolation Pilot Plant. Drums that are certified for shipment to the Waste Isolation Pilot Plant would be placed in interim storage on concrete pads in E-Area. Low-level radioactive waste containing concentrations of transuranic nuclides between 10 and 100 nanocuries (referred to as alpha-contaminated low-level radioactive waste) is managed like transuranic waste because its physical and chemical properties are similar and similar procedures would be used to determine its final disposition (DOE 1996c). The Waste Isolation Pilot Plant is expected to begin receiving waste from SRS in 2000 (DOE 1999b).

3.3.11.4 Low-Level Radioactive Waste

Both liquid and solid low-level radioactive waste are treated at SRS. Most aqueous low-level radioactive waste streams are sent to the F- and H-Area Effluent Treatment Facility and treated by filtration, reverse osmosis, and ion exchange to remove the radionuclide contaminants. After treatment, the effluent is discharged to Upper Three Runs. The treatment residuals are concentrated by evaporation and stored in the H-Area Tank Farm for eventual treatment in the Z-Area Saltstone Facility. In that facility, waste is immobilized with grout for onsite disposal (DOE 1996c).

After completion of a series of extensive readiness tests, the Consolidated Incinerator Facility began radioactive operations in 1997. The Consolidated Incinerator Facility is designed to incinerate both solid and liquid low-level radioactive waste, mixed waste, and hazardous waste (DOE 1999h).

Solid low-level radioactive waste is segregated into several categories to facilitate proper treatment, storage, and disposal. Solid low-level radioactive waste with a dose rate of less than 200 millirem per hour at 5 centimeters (2 inches) from an unshielded container is considered low-activity waste. If its dose rate is greater than 200 millirem per hour at 5 centimeters (2 inches), it is considered intermediate-activity waste. Intermediate-activity tritium waste is intermediate-activity waste with more than 10 curies of tritium per

container. Long-lived radioactive waste is contaminated with long-lived isotopes that exceed the waste acceptance criteria for onsite disposal (DOE 1996c).

Four basic types of vaults and buildings are used for storing the different waste categories: low-activity radioactive waste vaults, intermediate-level radioactive nontritium vaults, intermediate-level radioactive tritium vaults, and the long-lived radioactive waste storage building. The vaults are below-grade concrete structures, and the storage building is a metal building on a concrete pad (DOE 1996c).

Currently, DOE places low-activity low-level radioactive waste in carbon steel boxes and deposits them in the low-activity waste vaults in E-Area. Intermediate-activity low-level radioactive waste is packaged according to waste form and disposed of in the intermediate-level radioactive waste vaults in E-Area. Long-lived radioactive waste is stored in the Long-Lived Waste Storage Building in E-Area until treatment and disposal technologies are developed (DOE 1998a).

Saltstone generated in the solidification of low-level radioactive waste salts extracted from high-level radioactive waste is disposed of in the Z-Area Saltstone Vaults. Saltstone is solidified grout formed by mixing the low-level radioactive waste salt with cement, fly ash, and furnace slag. Saltstone is the highest volume of solid low-level radioactive waste disposed of at SRS. SRS disposal facilities are projected to meet solid low-level radioactive waste disposal requirements, including low-level radioactive waste from off site, for the next 20 years (DOE 1996c).

3.3.11.5 Mixed Waste

The Federal Facilities Compliance Agreement addresses SRS compliance with RCRA Land Disposal Restrictions. The Agreement requires DOE facilities storing mixed radioactive waste to develop site-specific treatment plans and to submit them for approval (DOE 1996c). The site treatment plan for mixed radioactive waste specifies treatment technologies or technology development schedules for all SRS mixed radioactive waste (DOE 1998a). SRS is allowed to continue to generate and store mixed radioactive waste, subject to Land Disposal Restrictions. Schedules to provide compliance through treatment in the Consolidated Incinerator Facility are included in the Federal Facilities Compliance Agreement (DOE 1996c).

The SRS mixed radioactive waste program consists primarily of safely storing waste until treatment and disposal facilities are available. Mixed waste is stored in the A-, E-, M-, N-, and S-Areas in various tanks and buildings. These facilities include burial ground solvent tanks, the M-Area Process Waste Interim Treatment/ Storage Facility, the Savannah River Technology Center Mixed Waste Storage Tanks, and the Defense Waste Processing Facility Organic Waste Storage Tank. These South Carolina Department of Health and Environmental Control-permitted facilities will remain in use until appropriate treatment and disposal is performed on the waste (DOE 1999h).

3.3.11.6 Hazardous Waste

Hazardous waste is accumulated at the generating facility for a maximum of 90 days, or stored in U.S. Department of Transportation-approved containers in three RCRA-permitted hazardous waste storage buildings and on three interim status storage pads in B- and N-Areas. Most of the waste is shipped off site to commercial RCRA-permitted treatment and disposal facilities using U.S. Department of Transportation-certified transporters. DOE plans to incinerate up to 9 percent of the hazardous waste (organic liquids, sludge, and debris) in the Consolidated Incinerator Facility (DOE 1996c). In 1995, 72 cubic meters (94 cubic yards) of hazardous waste were sent to onsite storage. Of this amount, 20 cubic meters (26 cubic yards) were shipped off site for commercial treatment or disposal (DOE 1999h).

3.3.11.7 Nonhazardous Waste

In 1994, the centralization and upgrading of the sanitary wastewater collection and treatment systems at SRS were completed. The program included the replacement of 14 (of 20) aging treatment facilities scattered across the site with a new 4,160 cubic meters (1.1 million gallons) per day central treatment facility, and connecting them with a new 29-kilometer (18-mile) sanitary sewer system. The central treatment facility treats sanitary wastewater by the extended aeration activated sludge process. The treatment facility separates the wastewater into two forms, clarified effluent and sludge. The liquid effluent is further treated by the nonchemical method of ultraviolet light disinfection to meet NPDES discharge limitations for the outfall to Fourmile Branch. The sludge is further treated to reduce pathogen levels to meet proposed land application criteria. The remaining sanitary wastewater treatment facilities are being upgraded as necessary by replacing existing chlorination treatment systems with nonchemical ultraviolet light disinfection systems to meet NPDES limitations (DOE 1996c).

SRS has privatized the collection, hauling, and disposal of its sanitary waste (DOE 1999h). SRS-generated solid sanitary waste is sent to the Three Rivers Landfill, a permitted disposal facility (DOE 2000). SRS disposes of other nonhazardous waste that consists of scrap metal, powerhouse ash, domestic sewage, scrap wood, construction debris, and used railroad ties in a variety of ways. Scrap metal is sold to salvage vendors for reclamation. Powerhouse ash and domestic sewage sludge are used for land reclamation. Scrap wood is burned on the site or chipped for mulch. Construction debris is used for erosion control. Railroad ties are shipped off site for disposal (DOE 1996c).

3.3.11.8 Waste Minimization

The total amount of waste generated and disposed of at SRS has been and continues to be reduced through the efforts of the pollution prevention and waste minimization program at the site. This program is designed to achieve continuous reduction of waste and pollutant releases to the maximum extent feasible and in accordance with regulatory requirements while fulfilling national security missions (DOE 1996c). The program focuses mainly on source reduction, recycling, and increasing employee participation in pollution prevention. For example, nonhazardous solid waste generation in 1995 was 32 percent below that of 1994, and the disposal volume of other solid waste, including radioactive and hazardous waste, was 38 percent below 1994 levels. In 1995, SRS achieved a 9 percent reduction in its radioactive waste generation volume compared with 1994. Total solid waste volumes have declined by more than 70 percent since 1991. Radioactive solid waste volumes have declined by about 63 percent, or more than 17,000 cubic meters (22,000 cubic yards), from 1991 through 1995. In 1995, more than 2,990 metric tons (3,300 tons) of nonradioactive materials were recycled at SRS, including 963 metric tons (1,062 tons) of paper and cardboard (DOE 1999h). The pollution prevention projects reduced the total amount of waste generated at SRS in 1997 by approximately 18,200 cubic meters (23,800 cubic yards) (DOE 1998b).

3.3.11.9 Preferred Waste Management Alternatives From the Final Waste Management Programmatic EIS and Associated Records of Decision

Preferred Alternatives from the Waste Management Programmatic EIS (DOE 1997a) are shown in **Table 3–26** for the four waste types analyzed in this EIS. Management of this waste could result in the construction of new waste management facilities at SRS and the closure of other facilities. Decisions on the various waste types were announced in a series of Records of Decision that have been issued on the Waste Management Programmatic EIS. The transuranic waste Record of Decision was issued on January 20, 1998 (63 FR 3629); the hazardous waste Record of Decision was issued on August 5, 1998 (63 FR 41810); the high-level radioactive waste Record of Decision on August 26, 1999 (64 FR 46661); and the low-level and mixed low-level radioactive waste Record of Decision on February 25, 2000 (61 FR 10061). The transuranic waste Record of Decision states, "...each of the Department's sites that currently has or will generate [sic] transuranic

waste will prepare and store its transuranic waste on site...." The hazardous waste Record of Decision states that most DOE sites will continue to use offsite facilities for the treatment and disposal of major portions of the nonwastewater hazardous waste, and Oak Ridge Reservation and SRS will continue to treat some of their own hazardous waste on site and in existing facilities where this is economically favorable. The high-level radioactive waste Record of Decision states that immobilized high-level radioactive waste will be stored at the site of generation. DOE decided in the Record of Decision for the management and disposal of low-level and mixed low-level radioactive waste to perform minimum treatment of low-level radioactive waste at all sites and continue, to the extent practicable, disposal of onsite low-level radioactive waste at INEEL, Los Alamos National Laboratory, the Oak Ridge Reservation, and SRS. For the management and disposal of mixed low-level radioactive waste, DOE decided to treat this waste at the Hanford site, INEEL, the Oak Ridge Reservation, and SRS, with disposal at the Hanford site and the Nevada Test Site. More detailed information concerning DOE's alternatives for the future configuration of waste management facilities at SRS is presented in the Waste Management Programmatic EIS and the hazardous, transuranic, high-level, and low-level and mixed low-level radioactive waste Records of Decision.

Table 3–26 Preferred SRS Waste Management Alternatives From the Waste Management Programmatic EIS and Associated Records of Decision

Waste Type	*Preferred Action*
High-level radioactive	DOE prefers onsite storage of SRS's immobilized high-level radioactive waste pending disposal in a geologic repository. [a]
Transuranic and mixed transuranic	DOE has decided that SRS should prepare and store its transuranic waste on site pending disposal at the Waste Isolation Pilot Plant. [b]
Low-level radioactive	DOE has decided to treat SRS low-level radioactive waste on site. SRS has been selected as one of the regional disposal sites for low-level radioactive waste. [c]
Mixed	DOE has decided to treat SRS mixed waste on site, including the possibility of treating mixed waste generated at other sites. SRS was not selected as one of the regional disposal sites for mixed waste. [c]
Hazardous	DOE has decided to use commercial and onsite SRS facilities for treatment of SRS nonwastewater hazardous waste, and to continue to use onsite facilities for wastewater hazardous waste. [d]

[a] From the Record of Decision for high-level radioactive waste (64 FR 46661).
[b] From the Record of Decision for transuranic waste (63 FR 3629).
[c] From the Record of Decision for low-level and mixed low-level radioactive waste (65 FR 10061).
[d] From the Record of Decision for hazardous waste (63 FR 41810).
Sources: DOE 1997a, 63 FR 3629, 63 FR 41810.

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Chapter 4

Environmental Consequences

4. ENVIRONMENTAL CONSEQUENCES

Chapter 4 describes the environmental consequences of the proposed action and alternatives to treat and manage sodium-bonded spent nuclear fuel. It begins with a general discussion of the expected environmental consequences; the product and waste forms that would be generated from the proposed action; and the methodology for assessing health effects from radiological and chemical effluent. It follows with a detailed description of the environmental consequences for the No Action and the reasonable alternatives. The chapter provides separate discussions on the environmental consequences of the intersite transportation of sodium-bonded spent nuclear fuel; the cumulative impacts at each of the proposed sites; and the programmatic considerations associated with the proposed action. The chapter concludes with a look at several issues under the proposed action, such as unavoidable, adverse environmental impacts; relationships between local, short-term uses of the environment and the enhancement of long-term productivity; and irretrievable commitments of resources.

4.1 OVERVIEW OF ENVIRONMENTAL CONSEQUENCES

This *Final Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel* (SBSNF EIS) is in compliance with Council on Environmental Quality regulations that require the affected environment of proposed Federal actions to be "interpreted comprehensively to include the natural and physical environment and the relationship of people with the environment" (40 CFR 1508.14).

The environmental consequence analysis focused on potentially affected areas. These areas are discussed in detail: air quality, water resources, socioeconomics, public and occupational health and safety (normal operations and accident conditions), environmental justice, waste management, and transportation. For the remaining areas (i.e., land resources, visual resources, noise, geology and soils, ecological resources, and cultural and paleontological resources), analyses show that the proposed treatment activities would have minimal or no impact at the candidate sites regardless of the alternatives being considered. This is because existing facilities within developed areas would be used; no new land disturbance would take place and proposed activities would be consistent with current operations. Since none of the alternatives involve construction other than internal building modifications for installing new equipment, the effects of these modifications on any of the resources would be negligible and are not evaluated in this chapter.

The specific assumptions associated with the impact analysis common to all alternatives are provided in the appendices. The results of the assessment of environmental consequences are presented in this chapter. More detailed descriptions of the development of the impacts for some resource areas are presented in Appendices E through H, as follows:

- Appendix E, Evaluation of Human Health Effects From Normal Operations
- Appendix F, Evaluation of Human Health Effects From Facility Accidents
- Appendix G, Evaluation of Human Health Effects From Overland Transportation
- Appendix H, Environmental Justice Analysis

4.1.1 Presentation of the Environmental Consequences

The primary areas of concern are products and waste, impacts on the public, and occupational health and safety associated with the various sodium-bonded spent nuclear fuel treatment processes. Additional areas and topics covered in Chapter 4 include the following:

- Air Quality
- Water Resources
- Environmental Justice
- Socioeconomics
- Waste Management
- Transportation Impacts
- Short-term versus Long-term Resource Commitments
- Irreversible and Irretrievable Resource Commitments
- Cumulative Impacts

Several kinds of impacts are not discussed in Chapter 4 because they would not occur, they would be extremely small, and/or they are covered by other analyses:

Land Use—The treatment and management of sodium-bonded spent nuclear fuel would not require the construction of new facilities on previously undisturbed land at Argonne National Laboratory-West (ANL-W) or the Savannah River Site (SRS).

Intrasite Transportation—The incident-free impacts of intrasite transportation would be limited to radiation exposure to workers loading and unloading trucks and are included in the overall worker dose values presented for each process. The accident risks are bounded by the site accident risk analysis. Strict site safety procedures and short travel distances would limit the impacts to workers.

Noise—Noise impacts at the management sites would be minor and would be limited to noise generated during operations. No offsite noise impacts are expected except for minor changes in traffic noise levels.

Ecological Resources—Because no new construction in undisturbed areas would be required for the treatment and management of sodium-bonded fuel, there would be no disturbance to terrestrial and aquatic habitats or wetlands. Thus, there would be no negative impacts from construction on terrestrial or aquatic plants or animals, including threatened and endangered species.

Scientific evidence indicates that limiting chronic radiation doses to below 0.1 rad per day to the more radiosensitive species in terrestrial ecosystems provides adequate protection for the population. In the aquatic environment, limiting the chronic dose rate to 1 rad or less per day to an individual member of the aquatic population would provide adequate protection for the population (IAEA 1992). Therefore, limiting chronic radiation doses to below 0.1 rad per day would not harm animal or plant populations. This is equivalent to a dose of 100 millirem per day from direct radiation (inhalation dose) to an individual. Compliance with U.S. Department of Energy (DOE) Order 5400.5 to limit the exposure of the most exposed member of the public (a hypothetical individual residing at the site boundary) to 100 millirem per year (i.e., about 0.3 millirem per day from all pathways) and to 10 millirem per year from the air pathway (40 CFR 61 dose limit) makes it highly probable that dose rates to plants and animals in the same area would be less than 0.1 rad per day. The maximum annual dose to the most exposed member of the public under any one of the alternatives analyzed would be a small fraction (about 0.2 percent) of 1 millirem. Therefore, no radiological damage to plant and animal populations would be expected to result from the sodium-bonded spent nuclear fuel treatment processes.

Chemicals emitted to the environment during routine processing activities are presented under each alternative. These releases are essentially independent of the process being performed. They are generated from operation of support facilities, such as operation of emergency diesel generators during testing and/or fuel burning for facility heating and power production. The quantities of releases attributable to the treatment of the sodium-bonded spent nuclear fuel would be very small fractions of the current releases from each management site.

The site environmental reports did not identify any measurable impacts on plants or animals because the amounts emitted are very low or the chemicals have little potential for causing negative effects.

For the reasons discussed above, no adverse impacts to ecological resources would be expected to occur due to DOE's treatment and management of sodium-bonded spent nuclear fuel.

Cultural and Paleontological Resources—No new facilities would be needed or constructed, therefore, there would be no impacts on cultural or paleontological resources.

Geology and Soils—No new facilities would be needed or constructed. Therefore, there would be no disturbance to either geologic or soil resources at the management sites. Hazards from large-scale geologic conditions were analyzed in detail in various DOE programmatic environmental impact statements (EISs) and site-specific facility safety analysis reports. The impacts from these hazards (e.g., earthquakes) on the management facilities and treatment processes are evaluated in this EIS.

4.1.2 Products and Waste

Generation and Disposition—All of the treatment processing alternatives presented in this EIS, except for direct disposal in high-integrity cans, would change the sodium-bonded spent nuclear fuel into other forms. Driver and blanket sodium-bonded spent nuclear fuel would be input—products and waste would be the output. The products and waste would be better suited for storage, transportation, and disposal or other disposition than the existing sodium-bonded fuel. The products and waste fall into several distinct categories:

- Materials to be managed as high-level radioactive waste would be generated at SRS and/or ANL-W. The final form would be ceramic, metallic, a melt and dilute product, or borosilicate glass inside stainless steel canisters. The production of ceramic, metallic, and melt and dilute products at ANL-W would result mainly from the transformation of spent nuclear fuel to a different form that would make the final product more stable and lead to an overall reduction in repository volume need. This waste would be stored at SRS and/or ANL-W until a geologic repository is ready to receive it.

- Transuranic waste refers to processed materials that contain alpha-emitting material (such as plutonium) with radioactivity concentrations above 100 nanocuries per gram of waste. Transuranic waste would be generated from all treatment technologies. This waste could be disposed of in the Waste Isolation Pilot Plant.

- The separated uranium resulting from the electrometallurgical treatment process at ANL-W would be made into solid metal ingots. The separated uranium resulting from processing the driver spent nuclear fuel would be made into low-enriched uranium ingots. The ingots would be more than 99.7 percent pure uranium; the balance of the material would be mainly zirconium (the alloy in the fuel) and trace quantities of fission products and actinides requiring additional purification before the uranium ingots could be used commercially. The uranium ingots would be stored in secure facilities along with other uranium already in storage at ANL-W until decisions are made about their disposition.

- Separated depleted uranium from plutonium-uranium extraction (PUREX) processing of declad and cleaned blanket spent nuclear fuel at SRS would be made into uranium oxides and stored in drums along with other depleted uranium at SRS (more than 27,000 metric tons of depleted uranium are stored currently at SRS). The 57 metric tons of depleted uranium that would be separated in this process would be a small fraction of what is stored currently.

- Separated plutonium resulting from PUREX processing of declad and cleaned blanket spent nuclear fuel at F-Canyon would be in a metallic form. The separated plutonium, less than 260 kilograms (572 pounds),

would be stored in secure facilities along with the plutonium already in storage (about 2.4 metric tons) at SRS. The plutonium would be disposed of in accordance with the Record of Decision (75 *FR* 1608) for the *Surplus Plutonium Disposition Final Environmental Impact Statement* (DOE 1999d).

- Low-level radioactive waste would be generated from all treatment technologies considered. This waste would be disposed of in existing facilities using routine procedures.

- Saltstone would be generated only at SRS. Saltstone is a form of concrete containing low levels of radioactivity and would be disposed of on site.

- Mixed waste would be generated at ANL-W and SRS. At ANL-W, mixed waste would be generated mainly from cadmium contamination, which would be present in one of the electrorefiners. It would consist of waste categorized as indirect process solid waste and would include discarded equipment and materials from decontamination operations. At SRS, liquid mixed waste would be generated from contamination by various chemicals in the dissolution and extraction facilities.

Waste Minimization—DOE would incorporate the best available practices into all the processing technologies at the two management sites to generate the smallest possible amount of waste. The DOE sites managing the sodium-bonded spent nuclear fuel would comply with DOE's waste minimization and pollution prevention goals. The following summarizes recent achievements in pollution prevention and waste minimization at ANL-W and SRS:

- ANL-W conducted pollution prevention projects in 1997 that reduced waste generation by an estimated 1,700 cubic meters (61,100 cubic feet) at a cost savings of $154,000. Radioactive waste generation in 1997 was reduced by 61 percent compared to 1993 baseline levels. Mixed waste generation was increased by 67 percent, hazardous waste generation was reduced by 44 percent, and sanitary waste generation was reduced by 32 percent compared to baseline levels. Fifty-six percent of sanitary waste was recycled in 1997. ANL-W affirmative procurement purchases are not tracked separately, and are included in the Idaho National Engineering and Environmental Laboratory (INEEL) totals. For INEEL, 72 percent of the materials purchased were U.S. Environmental Protection Agency (EPA)-designated recycled products (DOE 1998e).

- SRS conducted pollution prevention projects in 1997 that reduced waste generation by an estimated 18,200 cubic meters (644,000 cubic feet) at a cost savings of $18.5 million. Radioactive waste generation in 1997 was reduced by 57 percent compared to 1993 baseline levels. Mixed waste generation was increased by 115 percent, hazardous waste generation was reduced by 15 percent, and sanitary waste generation was reduced by 58 percent compared to baseline levels. Seventy-eight percent of sanitary waste was recycled in 1997, and 52 percent of the materials purchased under the affirmative procurement process were EPA-designated recycled products (DOE 1998e).

4.1.3 Radiological and Chemical Health Risk Estimates

The methodologies used to evaluate potential radiological and chemical health effects from operational effluent are described in Appendix E. This section provides information about the development and interpretation of the health risk estimates.

Radiological—The effect of radiation on people depends upon the kind of radiation exposure (alpha, beta, and neutron particles and gamma and x-rays), duration of exposure, 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."

An individual may be exposed to radiation from outside or 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 regulatory annual radiation dose limits to the maximally exposed offsite individual from total operations at a DOE site are 10 millirem from atmospheric pathways, 4 millirem from drinking water pathways, and 100 millirem from all pathways combined (DOE Order 5400.5 and 40 CFR Part 61, Subpart H). The potential doses associated with the normal operation of various treatment technologies and storage of sodium-bonded spent nuclear fuel would be very small fractions of these values, and total site doses would remain well within these DOE limits. For comparison, DOE estimates that the average individual in the United States receives a dose of approximately 360 millirem per year from all radiation sources combined, including natural and medical sources (see Appendix E, Section E.2.1, for details).

The collective or "population" dose to an exposed population is calculated by summing the estimated doses received by each member of the exposed population. The collective 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 person-rem (1,000 persons × 0.001 rem = 1 person-rem). The same population dose (1 person-rem) would result if 500 people each received a dose of 0.002 rem (500 persons × 0.002 rem = 1 person-rem).

Radiation can 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 from environmental and occupational radiation exposures (which are typically low doses) is the potential inducement of fatal cancers. This effect is referred to as "latent" cancer fatalities because the cancer may take years to develop and for death to occur.

In addition to latent cancer fatalities, other health effects could result from exposures to radiation. These effects include nonfatal cancers among the exposed population and genetic effects in subsequent generations. The dose-to-effect factors for fatal and nonfatal cancers are shown in **Table 4–1**. As indicated in this table, the nonfatal cancers and genetic effects are less probable consequences per unit of radiation exposure. For simplicity, this EIS presents estimated effects of radiation only in terms of latent cancer fatalities. Estimates of the total detriment (fatal cancers, nonfatal cancers, and genetic effects) due to radiation exposure may be obtained from the estimates of latent cancer fatalities presented in this EIS by multiplying by 1.4 for workers and by 1.46 for the general public.

The factors used in this EIS to relate a dose to its effect are 0.0004 lifetime probability of a latent cancer fatality per person-rem for workers and 0.0005 lifetime probability of a latent cancer fatality per person-rem for individuals among the general population. The latter factor is slightly higher because some individuals in the public, such as infants and children, are 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* (10 CFR 20). 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 higher doses and dose rates, the factors used to relate radiation doses to latent cancer fatalities are doubled. At much higher doses, prompt effects, rather than latent cancer fatalities risk, may be the primary concern.

Table 4–1 Risk of Latent Cancer Fatalities and Other Health Effects From Exposure to 1 Rem of Radiation[a]

Individual[b]	*Latent Cancer Fatalities*	*Nonfatal Cancers*	*Genetic Effects*	*Total Detriment*
Worker	0.0004	0.00008	0.00008	0.00056
Public	0.0005	0.0001	0.00013	0.00073

[a] When applied to an individual, units are lifetime probability of a latent cancer fatality per rem of radiation dose. When applied to a population of individuals, units are the excess number of cancers per person-rem of radiation dose. Genetic effects as used here apply to populations, not individuals.

[b] The difference between the worker risk and the general public risk is attributable to the fact that the general population includes more individuals in the more sensitive age group of less than 18 years of age.

Note: One rem equals 1,000 millirem.

Sources: NCRP 1993, ICRP 1991.

These factors are used to calculate the statistical expectations of the effects of exposing a population to radiation. For example, if 100,000 people each were exposed to a one-time radiation dose of 100 millirem (0.1 rem), the collective dose would be 10,000 person-rem. The exposed population then would be expected to experience 5 additional latent cancer fatalities from the radiation (10,000 person-rem × 0.0005 lifetime probability of a latent cancer fatality per person-rem = 5 latent cancer fatalities).

Sometimes calculations of the number of latent cancer fatalities associated with radiation exposure do not yield whole numbers and, especially in environmental applications, may yield numbers less than 1. For example, if 100,000 people each were exposed to a total dose of only 1 millirem (0.001 rem), the collective dose would be 100 person-rem, and the corresponding estimated number of excess latent cancer fatalities would be 0.05 (100,000 persons × 0.001 rem × 0.0005 latent cancer fatalities per person-rem = 0.05 latent cancer fatalities). The "0.05" means that there is one chance in 20 that the exposed population would experience one latent fatal cancer. In other words, the latent cancer fatality rate of 0.05 is the *expected* number of deaths that would result if the same exposure situation were applied to many different groups of 100,000 people. In most groups, nobody (0 people) would incur a latent cancer fatality from the 1 millirem dose each member would have received. In a small fraction of the groups, one latent cancer fatality would result; in exceptionally few groups, two or more latent cancer fatalities would occur. The average expected number of deaths for all the groups would be 0.05 latent cancer fatalities (just as the average of 0, 0, 0, and 1 is 1/4, or 0.25). The most likely outcome is 0 latent cancer fatalities.

The same concept is applied to estimate the effects of continuous radiation exposure to an individual member of the public. Consider the effects of an individual's exposure to a 360-millirem (0.36-rem) annual dose from all radiation sources (natural and medical). The probability that the individual would develop a latent fatal cancer from continuous exposure to this radiation over an average life of 72 years (presumed) is 0.013 (1 person × 0.36 rem per year × 72 years × 0.0005 latent cancer fatality risk per person rem = 0.013), or one chance in 77 that the individual would develop a fatal cancer from this radiation exposure.

The estimates of health effects from radiation doses used in this EIS are based on the linear no-threshold theory of radiation carcinogenesis, which postulates that all radiation doses, even those close to 0, are harmful. As explained in Appendix E, the numerical estimates of fatal cancers presented in this EIS were obtained from the nominal risk estimated for lifetime total cancer mortality, resulting from a dose of 0.1 gray (10 rad) (National Research Council 1990). Other methods of extrapolation to the low-dose region could yield higher or lower numerical estimates of fatal cancers. Studies of human populations exposed to low doses are inadequate to demonstrate the actual level of risk. There is a scientific uncertainty about cancer risk in the low-dose region below the range of epidemiologic observation, and the possibility of no risk cannot be excluded (CIRRPC 1992).

This EIS provides radiation dose estimates and probabilities of latent cancer fatalities (risks) for various receptors from management facility radiation exposure during normal operations and accident conditions. The receptors are defined as follows:

Worker – An individual actively participating and/or supporting the operation of the facility.

Noninvolved worker– An individual who is not involved in the operation of the facility. For estimating the impact, the individual is assumed to be 100 or more meters (330 or more feet) from the radioactive or chemical material release point.

Maximally exposed offsite individual – An individual member of the public assumed to be residing at the site boundary who could receive the maximum dose from radiation or hazardous chemicals.

Population – members of the general public residing within an 80-kilometer (50-mile) radius of the facility.

For incident-free (normal) operations, the EIS provides two sets of impacts (dose and risk): maximum annual and project total impacts for all alternatives. The maximum annual impacts result from simultaneous treatment of both driver and blanket sodium-bonded spent nuclear fuel in a given year, and the project total impacts represent the overall impacts from treatment of all sodium-bonded spent nuclear fuel. For accident conditions, the EIS provides both the consequence (dose) per accident and the associated risk.

Chemical—The potential impacts of exposure to hazardous chemicals released to the atmosphere as a result of the processing of sodium-bonded spent nuclear fuel were evaluated for incident-free operations and accident conditions at management facilities. Small amounts of hazardous and toxic chemical releases would be expected from incident-free operation of the treatment technology support facilities and equipment (e.g., auxiliary steam power house, diesel generators). The health effects from these releases were calculated for the maximally exposed offsite individual (an individual member of the public residing at the site boundary). The health effects evaluated in this analysis include excess latent cancer fatalities and chemical-specific noncancer health effects. The maximally exposed offsite individual was assumed to be located in the region with the highest estimated concentration. The health effects from releases of hazardous chemicals during accident conditions were evaluated in terms of comparison to Emergency Response Planning Guideline (ERPG) values. ERPG values are estimates of airborne concentration thresholds above which one can reasonably anticipate to observe adverse effects (see Appendix F, Section F.3.1.2, for more detail).

4.2 No Action Alternative

Under the No Action Alternative, the sodium-bonded spent nuclear fuel would not be treated (no sodium would be removed from the fuel). Under this alternative, two options were considered.

The EIS evaluates:

a. The impacts from the activities required to monitor and stabilize the sodium-bonded spent nuclear fuel as necessary for continued safe and secure storage at current locations, or until a new treatment technology, such as the glass material oxidation and dissolution system (GMODS) or plasma arc, is developed (see Section 2.6 for more details on GMODS and plasma arc technology development needs).

b. The impacts from direct disposal of sodium-bonded spent nuclear fuel in a geologic repository by packaging the fuel in high-integrity cans without sodium removal. At the present time, direct disposal of sodium-bonded spent nuclear fuel is precluded by DOE policy concerning acceptance of Resource Conservation and Recovery Act (RCRA)-designated mixed waste (which contains both hazardous and radioactive waste).

Under either option of the No Action Alternative, the EIS evaluates the impacts associated with activities required to clean and stabilize the waste materials generated during the Electrometallurgical Treatment Research and Demonstration Project at ANL-W. As part of this demonstration project, approximately 1.6 metric tons of heavy metal of Experimental Breeder Reactor -II (EBR-II) fuel consisting of about 1.2 metric tons of blanket spent nuclear fuel and 0.4 metric tons of driver spent nuclear fuel were processed (DOE 1996b). The waste materials generated in this project currently are being transformed into ceramic and metallic waste forms. This process will continue until all of this waste is transferred to ceramic and metallic waste forms. The remaining sodium-bonded spent nuclear fuel in the treatment facilities will be packaged and transferred to dry storage in the Radioactive Scrap and Waste Facility.

DOE also is transferring to dry storage all INEEL spent nuclear fuel, including the sodium-bonded spent nuclear fuel currently stored at Idaho Nuclear Technology and Engineering Center (INTEC) Building 603 (wet storage basin). During this transfer, each fuel can containing sodium-bonded fuel will be nondestructively examined to determine the fuel can condition and its suitability for storage. If any fuel can is found to be degraded, resulting in water in-leakage, it will be repackaged and transferred to ANL-W for stabilization and/or repackaging for storage. The fuel transfer activities are planned for completion by December 2000. The sodium-bonded spent nuclear fuel currently stored at INTEC Building 666 (wet storage basin) will remain in the basin until the planned defueling and facility closure in the year 2023. These fuel movement activities would be performed independently of the activities within this EIS.

About 5 metric tons of heavy metal of EBR-II blanket spent nuclear fuel contained in 107 storage cans currently stored at the Radioactive Scrap and Waste Facility do not meet the long-term confinement requirements. Under the No Action Alternative continued safe storage option. These storage cans would be brought to the Hot Fuel Examination Facility to be repackaged in more durable storage liners and would be returned to storage. These activities, along with the waste processing activities at ANL-W, would be completed in about two years after the necessary waste handling equipment is installed. The sodium-bonded spent nuclear fuel that currently is stored at INTEC would remain there, and sodium-bonded spent nuclear fuel transferred to Idaho in the future, as specified in the amended Record of Decision for the *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement* (Programmatic Spent Nuclear Fuel EIS) (61 FR 9441), also would be stored at INTEC. Consistent with the DOE-State of Idaho Settlement Agreement and Consent Order, all spent nuclear fuel would need to be transferred out of the State of Idaho by January 1, 2035. Under this option of the No Action Alternative, in the event that sodium-bonded spent nuclear fuel has not been treated before 2035, DOE would package the stored fuel at ANL-W and transfer it to the INEEL Dry Transfer facility. DOE also may decide to use the facilities at ANL-W to package the sodium-bonded spent nuclear fuel stored at INTEC. In the event that the sodium-bonded spent nuclear fuel has not been treated before 2035, the stored fuel would be removed from the State of Idaho by the year 2035. The environmental impacts of untreated sodium-bonded spent nuclear fuel removal would be evaluated in a separate National Environmental Policy Act (NEPA) document.

Under the No Action Alternative direct disposal option, all sodium-bonded spent nuclear fuel at INTEC would be transferred to ANL-W and repackaged in high-integrity cans in preparation for direct disposal. The activities associated with the preparation of sodium-bonded spent nuclear fuel for direct disposal would be similar to those needed to prepare the fuel for continued safe storage. The activities for direct disposal would occur sometime after those for the continued storage option. This is because a decision to directly dispose of

the sodium-bonded spent nuclear fuel in a geologic repository would be made only after it was determined that it would meet the repository acceptance criteria. Currently, there are no acceptance criteria for this fuel type. If direct disposal of the sodium-bonded spent nuclear fuel becomes possible, DOE would use the facilities at ANL-W to prepare all sodium-bonded spent nuclear fuel at the INEEL site. Preparation of driver spent nuclear fuel for direct disposal requires consideration of criticality safety, thereby limiting the amount of driver spent nuclear fuel that could be packaged in a canister. This would lead to larger repository volume needs per unit mass for driver fuel.

The activities in this option would include:

1. Repackaging 107 cans containing 5 metric tons of heavy metal of blanket spent nuclear fuel in the first two years (ending in 2003); see the continued storage option above.

2. Transferring sodium-bonded spent nuclear fuel currently stored at INTEC (Building 666 Basin and Building 603 Dry Storage) to ANL-W between 2003 and 2023. The 2023 date corresponds to the target date for closure of Building 666 Basin at INTEC. Under this assumption, the fuel in Building 666 Basin would be in wet storage for 23 years.

3. Repackaging the spent nuclear fuel at ANL-W in high-integrity cans to meet the target date for fuel transfer out of the State of Idaho (January 1, 2035). All sodium-bonded spent nuclear fuel would be transferred to the Hot Fuel Examination Facility for characterization and placement in high-integrity cans. The preparation and canning activities would be completed in about three years. The canned fuel would be stored temporarily at the Radioactive Scrap and Waste Facility. The stored fuel cans would be packaged in standardized canisters and transferred to the INEEL Dry Transfer Facility for packaging and shipment to the repository.

The environmental impacts for both options under the No Action Alternative are presented below. Where the impacts are different between the options, two sets of results are presented.

4.2.1 Air Quality

Nonradiological Gaseous Emissions

As explained in Appendix E, Section E.5.3.1, under the proposed action and either option of the No Action Alternative, small quantities of criteria pollutants and hazardous chemicals are generated from the operation of the emergency diesel generators supporting both the Fuel Conditioning Facility and the Hot Fuel Examination Facility at ANL-W. The emissions from these generators are independent of any of the treatment processes under the proposed action and the No Action Alternative addressed in this EIS.

Table 4–2 summarizes the concentrations of criteria and hazardous air pollutants. The concentrations are compared to their corresponding ambient air quality standards. Only those air pollutants that are expected and have ambient air quality standards are presented in the table. The emissions are generated from diesel generators currently in operation and are considered as part of the baseline concentration. No increases in emissions are expected under the No Action Alternative. Therefore, a Prevention of Significant Deterioration increment analysis was not required. In addition, the INEEL site is located in areas of attainment for the criteria pollutants; therefore, no conformity analysis is required.

Table 4–2 Nonradiological Air Quality Concentrations at the Site Boundary Under the No Action Alternative for Comparison With Ambient Air Quality Standards

	Averaging Period	*Most Stringent Standard or Guideline (micrograms per cubic meter)* [a]	*Maximum Incremental Concentration (micrograms per cubic meter)*
Criteria Pollutant			
Carbon monoxide	8 hours 1 hour	10,000 40,000	32.7 46.8
Nitrogen dioxide	Annual	100	2.8
PM_{10}	Annual 24 hours (interim) 24 hours (99th percentile over 3 years)	50 150 150	0.01 0.19 Not available
$PM_{2.5}$	3-year annual 24 hours (98th percentile over 3 years)	15 65	Not available Not available
Sulfur dioxide	Annual 24 hours 3 hours	80 365 1,300	0.45 11.50 25.80
Hazardous and Toxic Compounds			
1,3-Butadiene	Annual	0.0036	0.0000355
Acetaldehyde	Annual	0.45	0.0000226
Acrolein	24 hours	12.5	0.000181
Benzene	Annual	0.12	0.000694
Formaldehyde	Annual	0.077	0.0000709
Toluene	24 hours	18,750	0.00664
Xylene	24 hours	21,750	0.00447

PM_n = Particulate matter less than or equal to n microns in diameter.

[a] The standards for hazardous and toxic compounds apply only to increases in emissions from new or modified sources and are provided for information purposes only, as the concentrations from releases at ANL-W under all alternatives are not expected to increase.

Radiological Gaseous Emissions

Potential radiological releases from sodium-bonded spent nuclear fuel would be very small under both options of this alternative. Under both options, the spent nuclear fuel would remain stored in sealed canisters while at INEEL (i.e., INTEC or ANL-W) until 2035. However, degradation of sodium-bonded spent nuclear fuel or its enclosure (e.g., a sealed canister) during storage cannot be ruled out. It is expected that a small fraction of the fuel would degrade during storage, allowing its gaseous fission products to enter the storage canister. These fission gases would be released to the environment only if the sealed canister were to fail or be opened during fuel handling for examination and repackaging. As detailed in Appendix E, Section E.4.6, current experience at INTEC and ANL-W indicates very small fuel degradation problems during the storage period. It was estimated that, over 30 years of storage, about 1 percent of the fuel would be in degraded condition while in dry storage and about 3 percent of the fuel would fail while in wet storage. While in dry storage, there would be no releases of gaseous fission products to the environment. The fission gases would be released to the environment only during fuel repackaging. In wet storage, fuel canister degradation and resulting fuel failure would lead to releases of gaseous fission products. The estimated gaseous fission product releases during the entire period (over 35 years) of the No Action Alternative would be 51 curies of tritium oxides, 760 curies of krypton-85, and 0.000018 curies of iodine-129.

4.2.2 Water Resources

Surface Water

No surface water is used at ANL-W. Flood waters from the Big Lost River would not be expected to reach the facilities at ANL-W, as shown in Figure 3–3.

Nonradiological Liquid Effluent

There are no discharges to the surface waters at ANL-W, except for discharges of nonhazardous liquid waste to the sewage pond and to the industrial waste pond. Discharge waters to the Industrial Waste Pond or to the Sanitary Sewage Lagoons are not waters of the United States and are exempt from compliance under the National Pollutant Discharge Elimination System (NPDES). However, these are designated as waters of the State of Idaho and, as such, require compliance with State regulations that govern application of nonhazardous liquid waste (i.e., Land Application Permits). ANL-W has applied to the State of Idaho for Land Application Permits for the Industrial Waste Pond and Ditches and the Sanitary Waste Treatment Pond Land Application Area (DOE 1996a, DOE 1998c). ANL-W routinely monitors the effluent discharges to make sure they are within those limits identified in the Land Application Permits. Current operating and monitoring practices would continue for stormwater and liquid effluent discharges associated with facilities at ANL-W.

Radiological Liquid Effluent

No radiological liquid effluent would be discharged to the surface water.

Groundwater

Under either option of this alternative, there would be some reduction in groundwater consumption for domestic uses, since the number of workers at ANL-W is expected to decrease. The current water use at ANL-W is 188 million liters (49.6 million gallons) per year.

Nonradiological Liquid Effluent

For either option of this alternative, no nonradiological liquid effluent or waste would be discharged to groundwater.

Radiological Liquid Effluent

For either option of this alternative, no radiological liquid effluent would be discharged to groundwater.

4.2.3 Socioeconomics

Under either option of the No Action Alternative, there could be a reduction of approximately 350 workers at ANL-W if a treatment technology is not selected or the decision is delayed. If all of these workers were to leave the regional economic area, this could result in the loss of an additional 940 indirect jobs in the economic region. The total potential loss of about 1,290 jobs represents less than a 1 percent decrease in civilian employment in the regional economic area, which was estimated to be 150,403 in 1996 (DOE 1999d).

Since any reduction in the ANL-W labor force under the No Action Alternative would take place over time, combined with the fact that many of these workers could also support other missions at INEEL, the effects are expected to be gradual. By 2010, the contributory effect of this and the potential for beneficial effects from other industrial and economic sectors within the regional economic area would serve to reduce or mask any

effect on the regional economy. Neither option of the No Action Alternative, therefore, would result in any noticeable change in the existing regional economy, population and housing characteristics, or community services within the region of influence at ANL-W (see Section 3.2.8).

4.2.4 Public and Occupational Health and Safety

The assessments of potential radiological and chemical impacts associated with the No Action Alternative are presented in this section. Summaries of radiological impacts from normal operations are presented in Tables 4–3 and 4–4 for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in Tables 4–6 and 4–7. The impacts from hazardous chemical releases during accident conditions are presented in Table 4–8. Background information on the effects of radiation on human health and safety is presented in Section 4.1.3 and Appendix E, Section E.2.

4.2.4.1 Normal Operations

Radiological Impacts

Under either option of the No Action Alternative, radioactive releases from normal operations associated with spent nuclear fuel storage activities at ANL-W and INTEC would be small. The releases would occur from fuel degradation in wet storage and during fuel handling. Under both options, the same amount of gaseous radioactive material would be released. As explained in Appendix E, Section E.4.6, under both options, some fuel would be repackaged at the beginning in the first two years and all of the fuel would be repackaged by 2035 prior to shipment outside the INEEL site. The repackaging would occur over a three-year period. Releases would occur both from INTEC during wet storage and from ANL-W during fuel handling and repackaging operations. However, since INTEC is further away from the INEEL site boundary and major population centers compared to ANL-W, the releases were assumed to occur from ANL-W, thereby maximizing the impacts.

Calculated maximum annual and project total radiological impacts to the public are given in **Table 4–3**. The impacts are calculated for two types of receptors: the general public living within 80 kilometers (50 miles) of ANL-W in the year 2010, and a maximally exposed offsite individual (a member of the public assumed to be residing at the INEEL site boundary and receiving the maximum dose). To put the operational impacts into perspective, comparisons with impacts from natural background radiation also are included in the table. As shown in this table, the expected radiation doses to the maximally exposed offsite individual and the general public would be much smaller than the limit of 10 millirem per year limit set by the EPA (40 CFR 61) and DOE (DOE Order 5400.5).

The average worker dose (for ANL-W and INTEC workers) under the No Action Alternative was estimated to be similar to that currently experienced at ANL-W. Under both options, waste and fuel handling and repackaging activities would occur over a 5-year period, with standby operations for the remaining 30 years. One additional year also would be necessary to deactivate the facility. During fuel handling operations, the estimated annual total worker population dose would be 22 person-rem; during storage (standby) operations, it would be 2.2 person-rem; and during deactivation, it would be 33 person-rem, for a total of 209 person-rem over 35 years (see **Table 4–4**).

Table 4–3 Annual and Project Total Radiological Impacts to the Public From Operational Activities Under the No Action Alternative

Receptor	*Impacts*
Population Within 80 Kilometers (50 Miles) in the Year 2010	
Collective dose (person-rem per year) [a]	0.0015
Excess latent cancer fatalities (per year)	7.5×10^{-7}
Project total excess latent cancer fatalities [b]	6.5×10^{-6}
Maximally Exposed Offsite Individual	
Dose (millirem per year) [a]	0.00026
Percent of annual background [c]	0.000072
Latent cancer fatality risk (per year)	1.3×10^{-10}
Project total lifetime cancer fatality risk [b]	1.1×10^{-9}
Average Individual Within 80 Kilometers (50 Miles)	
Dose (millirem per year) [d]	6.2×10^{-6}
Latent cancer fatality risk	3.1×10^{-12}
Project total lifetime cancer fatality risk [b]	2.7×10^{-11}

[a] Annual maximum dose during normal operations.
[b] Total calculated risk over 35 years.
[c] The annual natural background radiation level at INEEL is about 360 millirem for the average individual (see Table 3–8); the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 person-rem.
[d] Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of ANL-W in the year 2010 (240,338).

Table 4–4 Annual and Project Total Radiological Impacts to Workers From Operational Activities Under the No Action Alternative

Receptor	*Impacts*
Worker [a]	
Average worker dose (millirem per year)	60
Average worker latent cancer fatality risk (project total over 35 years)	0.00084
Worker Population	
Collective dose (person-rem per year)	22
Excess latent cancer fatalities (per year)	0.0088
Project total dose (person-rem)	209
Project total excess latent cancer fatalities	0.084

[a] The regulatory dose limit for an individual worker is 5,000 millirem per year (10 CFR 835). However, the maximum annual dose to a worker would be kept below the DOE Administrative Control Level of 2,000 millirem per year, as established for all DOE activities in DOE Order N 441.1.

Source: ANL 1999.

As shown in Tables 4–3 and 4–4:

- The maximum annual dose to the maximally exposed offsite individual would be 0.00026 millirem, with an associated risk of developing a lifetime fatal cancer of 1.3×10^{-10} per year (or one chance in 7.7 billion that the individual would develop a fatal cancer per year of exposure).

- The collective maximum annual dose to the population within 80 kilometers (50 miles) of the storage facilities at ANL-W would be 0.0015 person-rem, with an associated 7.5×10^{-7} latent cancer fatalities per year (or one chance in 1.3 million that the population would experience a fatal cancer per year of exposure).

- The project total risk to the population within 80 kilometers (50 miles) of ANL-W would be 6.5×10^{-6} latent cancer fatalities (or one chance in 154,000 that the exposed population would experience a fatal cancer).

- The collective dose to facility workers would be 22 person-rem per year, with an associated 0.0088 latent cancer fatalities per year (or one chance in 113 that the workers would experience a fatal cancer per year of operation).

- The project total dose to facility workers would be 209 person-rem with an associated 0.084 latent fatal cancers (or one chance in 12 that the exposed workers would experience a fatal cancer).

These results indicate that no increase in the expected number of fatal cancers among the various affected populations would result from the activities under this alterative.

Hazardous Chemical Impacts

The hazardous chemical impacts to the public residing at the INEEL site boundary from the operational activities at ANL-W under either option of this alternative are summarized in **Table 4–5**. Appendix E, Section E.5, provides details on the model used and results obtained. The results (presented in Table 4–5) indicate that no adverse toxic health or cancer effects would be expected from exposure to hazardous chemicals released under this alternative. The existing chemical environment is presented in Section 3.2.10.2.

Table 4–5 Hazardous Chemical Impacts to the Public From Operational Activities Under the No Action Alternative

Chemical	*Maximum Annual Concentration (milligrams per cubic meter)*	*Hazard Quotient (noncarcinogenic chemicals)*	*Cancer Risk (carcinogenic chemicals)*
1,3-Butadiene	3.6×10^{-8}	None	9.9×10^{-9}
Acetaldehyde	2.3×10^{-8}	2.5×10^{-6}	5.0×10^{-11}
Acrolein	7.1×10^{-9}	0.00035	None
Benzene	6.9×10^{-7}	None	5.4×10^{-9}
Formaldehyde	7.1×10^{-8}	None	9.2×10^{-10}
Toluene	2.5×10^{-7}	6.2×10^{-7}	None
Hazard Index		0.00036	Not applicable

4.2.4.2 Facility Accidents

The potential radiological impacts to the public and noninvolved onsite workers due to accidents are summarized in this section. The detailed analysis of facility accidents, with the associated assumptions, is presented in Appendix F. The detailed analysis considered a wide spectrum of potential accident scenarios, including fire, spills, criticality, an earthquake, and an aircraft crash.

Under either option of the No Action Alternative, spent nuclear fuel transfer and waste processing activities associated with cleaning and stabilizing the waste materials generated during the Electrometallurgic Treatment

Research and Demonstration Project at ANL-W would be performed. These activities would have the potential to involve accident scenarios similar to those evaluated for Alternative 1 as presented in Section 4.3.4.2. However, the consequences associated with these accident scenarios would be lower because of the limited quantities of waste to be stabilized. Accidents associated with spent nuclear fuel transfer activities also could occur during fuel removal from the Radioactive Scrap and Waste Facility and packaging for offsite shipment to a repository. These accidents would lead to consequences similar to those evaluated for Alternative 1 as presented in Section 4.3.4.2. It is estimated that the spent nuclear fuel transfer and waste stabilization activities would occur over a two-year period. Fuel handling and repackaging for offsite shipment would occur over a three-year period.

No reasonably foreseeable accident scenarios could be identified that would impact sodium-bonded spent nuclear fuel in dry storage at the Radioactive Scrap and Waste Facility or in wet or dry storage at INTEC. In storage, the sodium-bonded spent nuclear fuel is in a safe and stable configuration. Generally, the only activity associated with the stored spent nuclear fuel is monitoring of the fuel and the storage facility. While in storage, activities that could lead to accidents (movement, repackaging, or processing of the spent nuclear fuel) are not performed. However, approximately 1.2 metric tons of sodium-bonded spent nuclear fuel currently in wet storage in Basin 603 at INTEC would be transferred to dry storage facilities at INTEC. Handling accidents could occur during transfer activities at INTEC similar to the accident scenarios evaluated for ANL-W. Because INTEC is further away from the INEEL site boundary and major population centers compared to ANL-W, the health impacts from accidents at INTEC would be less than those from similar accidents at ANL-W.

Table 4–6 presents the frequencies and consequences of the postulated set of accidents to the maximally exposed offsite individual, the offsite population residing within 80 kilometers (50 miles) of the facility, and a noninvolved worker. The dose to the maximally exposed offsite individual was calculated for the 95th percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50th 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 that produce higher calculated exposures, and is defined as that condition that is not exceeded more than 5 percent of the time. DOE did not quantitatively estimate the involved worker dose due to accidents. The consequences to involved workers are qualitatively assessed. This approach is used for the following two reasons: (1) no adequate method exists for calculating meaningful consequences at or near the location where the accident occurs, and (2) safety assurance for facility workers is demonstrated by both the workers' training and by the establishment of an Occupational Safety and Health Administration (OSHA) process safety management system (29 CFR 1910.119). In any accident scenario, the individuals most likely to be injured are the involved workers. The risk to these workers would be due to both radiological and nonradiological effects. In a fire, the involved workers could be exposed to airborne radioactive material, in addition to the smoke and heat of the fire. In an explosion, there could be flying debris and containment barriers could be broken, exposing workers to airborne radioactive material. Most spills would not have a major effect on involved workers because they would clean up the spill wearing protective clothing and respirators as necessary. An accidental criticality could expose involved workers to large doses of prompt penetrating radiation, which could cause death in a short period of time. An earthquake accident could present a very severe nonradiological effect to the involved workers. For example, in a beyond-design-basis earthquake, the workers are likely to be hurt or could be killed from the collapse of the building before they could be evacuated (see Appendix F, Section F.2.2.2, for more detail). The accident risks are summarized in **Table 4–7**.

Table 4–6 Accident Frequency and Consequences Under the No Action Alternative

Accident [a]	Frequency (event per year)	Maximally Exposed Offsite Individual		Population Within 80 Kilometers (50 Miles)		Noninvolved Worker	
		Dose (millirem)	Latent Cancer Fatality Risk [b]	Dose (person-rem)	Excess Latent Cancer Fatalities [c]	Dose (millirem)	Latent Cancer Fatality Risk [b]
Salt powder spill in the Hot Fuel Examination Facility cell [d]	0.01	0.00046	2.3×10^{-10}	0.000098	4.9×10^{-8}	4.7×10^{-7}	1.9×10^{-13}
Cask drop during spent nuclear fuel transfer	0.01	0.03	1.5×10^{-8}	0.0035	1.7×10^{-6}	0.00084	3.4×10^{-10}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Design-basis earthquake	0.008	12	6.0×10^{-6}	1.4	0.00070	4.7	1.9×10^{-6}
Salt transfer drop	1×10^{-7}	0.19	9.5×10^{-8}	0.022	0.000011	0.073	2.9×10^{-8}
Beyond-design-basis earthquake	0.00001	96	0.000048	11	0.0055	37	0.000015

[a] Only accidents involving EBR-II driver spent nuclear fuel, which maximizes the consequences, are presented.
[b] Increased likelihood of a latent cancer fatality.
[c] Increased number of latent cancer fatalities.
[d] The salt powder spill was assumed to have similar characteristics to those evaluated under Alternative 1. The radionuclide concentration in this salt would be about one-third of those generated in Alternative 1.

Table 4–7 Annual Cancer Risks Due to Accidents Under the No Action Alternative

Accident	Maximally Exposed Offsite Individual [a]	Population Within 80 Kilometers (50 Miles) [b]	Noninvolved Worker [a]
Salt powder spill in Hot Fuel Examination Facility cell	2.3×10^{-12}	4.9×10^{-10}	1.9×10^{-15}
Cask drop during spent nuclear fuel transfer	1.5×10^{-10}	1.7×10^{-8}	3.4×10^{-12}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Design-basis earthquake	4.8×10^{-8}	5.6×10^{-6}	1.5×10^{-8}
Salt transfer drop	9.5×10^{-15}	1.1×10^{-12}	2.9×10^{-15}
Beyond-design-basis earthquake	4.8×10^{-10}	5.5×10^{-8}	1.5×10^{-10}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

For accidents at ANL-W, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be 4.8×10^{-8} per year (or one chance in 20.8 million that the individual would develop a fatal cancer per year of operation) and 1.5×10^{-8} per year (or one chance in 66.7 million that the individual would develop a fatal cancer per year of operation), respectively. The increased number of latent cancer fatalities in the surrounding population would be 5.6×10^{-6} per year (or one chance in 178,600 that the population would experience a fatal cancer per year of operation).

Hazardous Chemical Impacts

Nonradiological hazardous chemical impacts are evaluated in terms of comparison to ERPGs. ERPG values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects (see Appendix F, Section F.3.1.2, for details).

The nonradiological (hazardous chemical) impacts of potential facility accidents associated with either option of the No Action Alternative are summarized in **Table 4–8**.

Table 4–8 Hazardous Chemical Accident Impacts Under the No Action Alternative

Accident	*Frequency (Event Per Year)*	*Receptor*	*Exposure*
Uranium handling accident	0.01	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Design-basis earthquake	0.0002	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Beyond-design-basis earthquake	0.00001	Noninvolved worker	Cadmium: less than ERPG-1 Uranium: less than ERPG-1
		Maximally exposed offsite individual	Cadmium: less than ERPG-1 Uranium: less than ERPG-1

ERPG = Emergency Response Planning Guideline.

4.2.5 Environmental Justice

As discussed in Appendix H, Executive Order 12898 directs Federal agencies to address disproportionately high and adverse health or environmental effects of alternatives on minority or low-income populations.

Analyses of normal operations and accident conditions presented in Section 4.2.4 show the risk of latent cancer fatalities to the public residing within 80 kilometers (50 miles) of the spent nuclear fuel storage facilities at ANL-W and INTEC to be much lower than 1. Therefore, there would be no disproportionately high and adverse consequences for any particular group within the general population, including minority or low-income populations, beyond the effects of existing and future activities that are independent of the proposed action.

4.2.6 Waste Management

Various types of waste would be generated as a result of sodium-bonded spent nuclear fuel storage activities at ANL-W, including transuranic waste, low-level radioactive waste, mixed waste, hazardous, and nonhazardous waste. In addition, during the first two years of operation under either option of this alternative, ANL-W would continue to generate high-level radioactive waste as the Electrometallurgical Treatment Research and Demonstration Project cladding hull waste and electrorefiner salt are stabilized to metallic and ceramic high-level radioactive waste forms for ultimate disposal. **Table 4–9** shows the anticipated categorization of these waste types and their expected interim storage and final disposal locations. The quantities of ceramic and metallic waste forms generated, along with other generated waste, are presented in **Table 4–10**. The values in Table 4–10 are for disposal (solid waste) and account for volume reduction.

Direct Process Waste

Under either option of the No Action Alternative, small amounts of metallic and ceramic high-level radioactive waste would be produced at ANL-W as a result of the completion of the demonstration project. The salt removed from the electrorefiners would contain the majority of fission products and transuranics from the spent nuclear fuel. This removed salt would be packaged and transferred to the Hot Fuel Examination Facility for processing into ceramic waste. The metallic waste form would consist primarily of stainless steel cladding hulls containing the noble metal fission products. Both the ceramic and metallic waste would be categorized as high-level radioactive waste. The volumes of waste forms provided in Table 4–10 are for the standardized canisters required for disposal of these materials.

Table 4–9 Waste Material Categories at INEEL and Interim and Final Locations

Waste Stream	*Category*	*Interim Storage Location*	*Final Disposal Location*
Process Waste			
Fuel hardware	Low-level radioactive waste	None	Radioactive Waste Management Complex
Metallic waste form	High-level radioactive waste	Radioactive Scrap and Waste Facility	Geologic repository
Ceramic waste form	High-level radioactive waste	Radioactive Scrap and Waste Facility	Geologic repository
Other Associated Process Waste			
Less than 10 nanocuries per gram transuranic waste [a]	Low-level radioactive waste	None	Radioactive Waste Management Complex
Greater than 100 nanocuries per gram transuranic waste	Transuranic waste	Radioactive Waste Management Complex	Waste Isolation Pilot Plant
Cadmium-contaminated	Mixed waste	Radioactive Scrap and Waste Facility	Waste Isolation Pilot Plant or Radioactive Waste Management Complex after treatment
Nonradioactive	Sanitary waste	None	INEEL landfill
Deactivation Waste			
Electrorefiner cadmium	Mixed waste	Radioactive Scrap and Waste Facility	Waste Isolation Pilot Plant
Equipment less than 10 nanocuries per gram transuranic waste [a]	Low-level radioactive waste	None	Radioactive Waste Management Complex
Equipment greater than 100 nanocuries per gram transuranic waste	Transuranic waste	Radioactive Waste Management Complex	Waste Isolation Pilot Plant
Cadmium-contaminated	Mixed waste	Radioactive Scrap and Waste Facility	Waste Isolation Pilot Plant or Radioactive Waste Management Complex

[a] As noted in Section 3.2.11.3, the Radioactive Waste Management Complex cannot be used for the disposal of the alpha low-level radioactive waste (between 10 and 100 nanocuries per gram). Waste in this category may be treated by the Advanced Mixed Waste Treatment Project and then disposed of at the Waste Isolation Pilot Plant.

The metallic and ceramic high-level radioactive waste generated as a result of the demonstration project at ANL-W would be stored temporarily at the Radioactive Scrap and Waste Facility at ANL-W in a manner that allows retrieval for future disposal. The Radioactive Scrap and Waste Facility was designed and constructed for temporary storage of this type of waste, and shielding will be provided by a combination of: (1) steel storage liners in which the waste would be stored and (2) by the soil surrounding the liners. When a geologic repository is available, the waste cans containing the metallic and ceramic high-level radioactive waste would be removed from storage, packaged in standardized canisters, shipped to the INEEL Dry Transfer Facility, and prepared for shipment to the repository. If direct disposal of sodium-bonded spent nuclear fuel becomes acceptable, the sodium-bonded spent nuclear fuel at INTEC would be transferred to ANL-W for repackaging, along with other fuel at ANL-W. The packaged canisters would be transferred to the INEEL Dry Transfer Facility for shipment off site to a repository.

Table 4–10 Amounts of Waste Generated Under the No Action Alternative[a]

Waste Stream	Waste Quantities	
	Volume (cubic meters)	Mass (kilograms)
Direct Process Waste		
Fuel assembly hardware (low-level radioactive waste)	0	0
High-level radioactive ceramic waste	9.4 (15 canisters)[b]	14,000
High-level radioactive metallic waste	0.6 (1 canister)[b]	460
Spent nuclear fuel	142 (355 canisters)[b]	72,000
Other Associated Process Waste		
Low-level radioactive waste	792	161,000
Transuranic waste	10.5	4,000
Mixed waste	40	21,500
Sanitary waste	2,500	867,000
Deactivation Waste		
Low-level radioactive waste	112	38,000
Transuranic waste	1.6	853
Mixed waste	3	2,100

[a] These waste generation estimates are through the year 2035. This is the date by which materials of this type are required to be out of the State of Idaho.
[b] Standardized canisters.
Source: ANL 1999.

Other Associated Process Low-Level Radioactive Waste

Low-level radioactive waste would be generated during conversion of demonstration high-level radioactive waste into suitable forms for the repository, as well as from other ongoing activities, including keeping a hot cell facility operational to handle unforeseen problems while storing the sodium-bonded spent nuclear fuel at INTEC or in the Radioactive Scrap and Waste Facility. In addition, low-level radioactive waste would be generated from activities in the Fuel Conditioning Facility and the Hot Fuel Examination Facility (e.g., equipment decontamination and repair), as well as in other facilities at ANL-W (e.g., analytical laboratory activities). Material in this waste stream has been generated and routinely handled at ANL-W for many years.

The volume of low-level radioactive waste resulting from either option of the No Action Alternative activities at ANL-W that would require disposal (after volume reduction) would be a maximum of about 50 cubic meters (1,766 cubic feet) per year during processing activities, and approximately 17 cubic meters (600 cubic feet) per year during the remaining years. This maximum volume represents a small fraction (approximately 1 percent) of the total annual volume of low-level radioactive waste currently being disposed of at the Radioactive Waste Management Complex. The total of 792 cubic meters (28,000 cubic feet) of low-level radioactive waste generated during either option represents approximately 0.7 percent of the total Radioactive Waste Management Complex disposal inventory.

Other Associated Process Transuranic Waste

Transuranic waste would be generated at ANL-W under either option of the No Action Alternative from decontamination activities for repair and maintenance of items, and miscellaneous work associated with

demonstration fuel processing or other activities. Transuranic waste would be generated primarily from activities conducted in gloveboxes and hot cells at ANL-W.

For the No Action Alternative, the volume of transuranic waste generated at ANL-W would amount to a maximum of approximately 1 cubic meter (35 cubic feet) per year during processing activities, and approximately 0.2 cubic meters (7 cubic feet) per year during the remaining years. This maximum volume is approximately 0.002 percent of the volume of transuranic waste in retrievable storage at the Radioactive Waste Management Complex. The total volume of incidental transuranic waste generated under either option is approximately 10.5 cubic meters (370 cubic feet), which is 0.006 percent of the estimated total volume of transuranic waste to be emplaced at the Waste Isolation Pilot Plant.

Other Associated Process Sanitary Waste

Sanitary waste, which is nonradioactive and nonhazardous solid waste, would continue to be generated under either option of the No Action Alternative. This waste would be typical of industrial operations and would be disposed of at the INEEL landfill. Based on an estimated eventual INEEL landfill volume of 3×10^6 cubic meters (106 million cubic feet), the total volume of sanitary waste generated and disposed of under this alternative is approximately 0.1 percent of the INEEL landfill volume.

Other Associated Process Mixed Waste

Mixed waste would be generated primarily from the disposal of any cadmium-contaminated equipment or clean-up material and the analysis of cadmium samples. At ANL-W, mixed waste would be handled according to ANL-W procedures that require limited accumulation at the point of generation. Interim storage of this waste would be at the Radioactive Scrap and Waste Facility prior to eventual disposal. The Radioactive Scrap and Waste Facility is a permitted mixed waste storage facility for these materials. The mixed waste streams that contribute to the overall mixed waste generated at ANL-W have been identified in the INEEL Site Treatment Plan (DOE 1995b).

Deactivation Waste

A variety of waste would be generated as part of deactivation activities at ANL-W. This would include process equipment and process material such as cadmium in one of the electrorefiners. Generated waste categories would include low-level radioactive waste, transuranic waste, and mixed waste. This waste would be categorized and disposed of according to DOE Orders and ANL-W radioactive waste management procedures, as described above for each waste category.

The largest volume of deactivation waste under either option of the No Action Alternative would be low-level radioactive waste, generated as a result of equipment dismantling and disposal. Components that would require disposal include the existing electrorefiners and hot isostatic press, as well as other processing components. Decontamination of these components would generate additional mixed, transuranic, and low-level radioactive waste that would require management. The deactivation waste volume would be generated over a period of one year. The total deactivation waste represents approximately 14 percent over the total associated process waste (excluding sanitary waste) requiring disposal.

4.3 Alternative 1: Electrometallurgically Treat Blanket and Driver Fuel at ANL-W

Under this alternative, the sodium-bonded spent nuclear fuel would be treated at ANL-W using the electrometallurgical process, described in Appendix C. The various process steps in this technology are performed at the Fuel Conditioning Facility and the Hot Fuel Examination Facility hot (air or argon) cells. The processes at the Fuel Conditioning Facility include: fuel chopping, electrorefining, cathode processing,

and metal casting (see Appendix C for details on each processing step). These processes would separate the uranium from the fission products. Separated uranium is not considered a waste. The separated uranium would be made into low-enriched uranium ingots, and the metallic sodium would be oxidized in the electrorefiner lithium-potassium salt and removed along with the fission products as high-level radioactive waste. The salts from the electrorefiner then would be solidified and sent to the Hot Fuel Examination Facility for further processing. The processes at the Hot Fuel Examination Facility include waste treatment, metallic melting, and high-level radioactive waste production. These processes would produce two waste forms—a ceramic waste form consisting of fission products and transuranic elements and a metallic waste form consisting of noble metal fission products and cladding hulls from the spent nuclear fuel. The low-enriched uranium metal ingots would be stored at the Zero Power Physics Reactor Material Storage Building. The ceramic and metallic waste forms would be temporarily stored at the Radioactive Scrap and Waste Facility pending packaging for disposition in a geologic repository.

The electrometallurgical process at ANL-W facilities would treat about 5 metric tons of heavy metal of sodium-bonded spent nuclear fuel per year. Appendix E, Section E.4.1, provides details on the process duration and the amount of blanket and driver spent nuclear fuel treated annually. The treatment of blanket and driver spent nuclear fuel under this alternative could start as early as 2000 and could be completed by 2012. In addition, a full year of operations would be needed to deactivate the processing equipment and establish an industrially safe configuration.

4.3.1 Air Quality

Nonradiological Gaseous Emissions

As explained in Appendix E, Section E.5.3.1, under all alternatives, small quantities of criteria and hazardous chemicals are generated from the operation of the emergency diesel generators supporting both the Fuel Conditioning Facility and the Hot Fuel Examination Facility at ANL-W. The emissions from these generators are independent of any of the treatment processes addressed in this EIS. In addition, the electrometallurgical treatment of driver fuel under Alternatives 1 through 5 would release small quantities of cadmium. This release would occur as an elevated release from the Fuel Conditioning Facility stack.

Table 4–11 summarizes the concentrations of criteria and hazardous air pollutants. The concentrations are compared to their corresponding ambient air quality standards. Only those air pollutants that are expected and have ambient air quality standards are presented in the table. The emissions are generated from diesel generators currently in operation and are considered as part of the baseline concentration. No increases in emissions are expected under this alternative. Therefore, a Prevention of Significant Deterioration increment analysis was not required. In addition, the INEEL site is located in areas of attainment for criteria pollutants; therefore, no conformity analysis is required.

Radiological Gaseous Emissions

Krypton-85 and elemental tritium are the most prevalent radioactive gaseous fission products that would be released to the argon cell at the Fuel Conditioning Facility during fuel element chopping and electrorefining processes. The released tritium in the cell would not be oxidized due to a very low presence of oxygen and humidity in the argon cell, and releases of tritium to the atmosphere would be in the elemental form. The oxidation of elemental tritium to tritium oxide (HTO or T_2O) has been shown to occur slowly in the environment, and in the long term, about 1 percent of tritium would be oxidized (see Appendix E, Section E.4.1, for more details). The argon cell also contains an equilibrium concentration of other radionuclide isotopes. Appendix E, Section E.4.1, provides a list of various isotopes that are present in the argon cell in nanocuries (10^{-9} curies) and are released to the environment through the facility stack, along with krypton and elemental tritium. The maximum release of radioactive gaseous emissions would occur during

the first six years of the electrometallurgical treatment process, when a combination of EBR-II blanket and driver spent nuclear fuel elements would be processed. During these six years, about 0.6 metric tons of heavy metal of driver spent nuclear fuel and about 4.4 metric tons of heavy metal of blanket spent nuclear fuel would be processed annually. The combined process would release about 11,600 curies of krypton-85 and 770 curies of elemental tritium annually. After six years and until the end of the processing period, the release rate would drop significantly. During this period, only Fermi-1 blanket spent nuclear fuel, with an annual release of about 0.4 curies of elemental tritium and 3.3 curies of krypton-85, would be processed. The radiological exposures to the public and workers from these emissions are presented in detail in Appendix E, Section E.4.1, and are summarized in Section 4.3.4.

Table 4–11 Nonradiological Air Quality Concentrations at the Site Boundary Under Alternative 1 at ANL-W for Comparison With Ambient Air Quality Standards

	Averaging Period	*Most Stringent Standard or Guideline (micrograms per cubic meter)* [a]	*Maximum Incremental Concentration (micrograms per cubic meter)*
Criteria Pollutant			
Carbon monoxide	8 hours 1 hour	10,000 40,000	32.70 46.80
Nitrogen dioxide	Annual	100	2.79
PM_{10}	Annual 24 hours (interim) 24 hours (99th percentile over 3 years)	50 150 150	0.01 0.19 Not available
$PM_{2.5}$	3-year annual 24 hours (98th percentile over 3 years)	15 65	Not available Not available
Sulfur dioxide	Annual 24 hours 3 hours	80 365 1,300	0.45 11.50 25.80
Hazardous and Toxic Compounds			
1,3-Butadiene	Annual	0.0036	0.0000355
Acetaldehyde	Annual	0.45	0.0000226
Acrolein	24 hours	12.5	0.000181
Benzene	Annual	0.12	0.000694
Cadmium	Annual	0.00056	3.58×10^{-10}
Formaldehyde	Annual	0.077	0.0000709
Toluene	24 hours	18,750	0.00664
Xylene	24 hours	21,750	0.00447

PM_n = Particulate matter less than or equal to n microns in diameter.

[a] The standards for hazardous and toxic compounds apply only to increases in emissions from new or modified sources and are provided for information purposes only, as concentrations from releases at ANL-W under all alternatives are not expected to increase.

4.3.2 Water Resources

Surface Water

No surface water is used at ANL-W. Flood waters from the Big Lost River would not be expected to reach the facilities at ANL-W, as shown in Figure 3–3.

Nonradiological Liquid Effluent

There are no discharges to the surface waters at ANL-W, except for discharges of nonhazardous liquid waste to the sewage pond and to the industrial waste pond. Big Lost River, Little Lost River, and Birch Creek would not be impacted by activities associated with the electrometallurgical treatment processes. Current operating and monitoring practices would continue for stormwater and liquid effluent discharges associated with facilities at ANL-W (see Section 4.2.2).

During fuel treatment and associated activities, some hazardous materials may be used inside buildings. To prevent potential releases to surface or subsurface waters resulting from spills of hazardous materials used in buildings, these facilities are designed, constructed, and maintained to contain these materials. Double-contained pipes, leak detection, and secondary containment of tanks are some of the features used to prevent hazardous material releases to the environment. Following existing written procedures, spill containment and cleanup equipment is present in areas where hazardous materials are stored or used (DOE 1996b).

Radiological Liquid Effluent

No radiological liquid effluent or waste generated by the electrometallurgical treatment process would be discharged to surface water.

Groundwater

Under this alternative at ANL-W, there would be little change in groundwater consumption for domestic use since there is little change expected in the number of workers. The current water use at ANL-W is 188 million liters (49.6 million gallons) per year.

Nonradiological Liquid Effluent

No nonradiological liquid effluent generated by the electrometallurgical treatment process would be discharged to groundwater.

Radiological Liquid Effluent

No radiological liquid effluent generated by the electrometallurgical treatment process would be discharged to groundwater.

4.3.3 Socioeconomics

Under this alternative, the existing facilities at ANL-W would remain operational. No new employment or in-migration of workers would be required. Thus, there would be no additional impacts on the socioeconomic conditions in the region around ANL-W and INEEL.

4.3.4 Public and Occupational Health and Safety

The assessments of potential incremental radiological and chemical impacts associated with this alternative are presented in this section. Summaries of radiological impacts from normal operations are presented in Tables 4–12 and 4–13 for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in Tables 4–15 and 4–16. The impacts from hazardous chemical releases during accident conditions are presented in Table 4–17. Background information on the effects of radiation on human health and safety is presented in Appendix E, Section E.2.

4.3.4.1 Normal Operations

Radiological Impacts

Under this alternative, radioactive releases would occur during fuel chopping and from the operation of electrorefiners. Both of these activities are performed in the Fuel Conditioning Facility argon cell. Appendix E, Sections E.3 and E.4.1, details the method and assumptions used for calculating the impacts of normal operational radiological releases on the public health and safety. The maximum annual dose to the public would result from treating 0.6 metric tons of heavy metal of EBR-II driver spent nuclear fuel and 4.4 metric tons of heavy metal of EBR-II blanket spent nuclear fuel. This combination of fuel treatment would continue for 6 years, after which only Fermi-1 blanket spent nuclear fuel with a very low radioactivity content would be treated. Overall, it would require 13 years to treat all the sodium-bonded fuel (see Appendix E, Section E.4.1, for details).

Calculated maximum annual and project total radiological impacts to the public from operational activities under this alternative are given in **Table 4–12**. The maximum dose to the public would occur during the first six years of operation. The annual dose to the public during Fermi-1 blanket spent nuclear fuel treatment would be very small (see Appendix E, Table E–8). The impacts are calculated for two types of receptors: the general public living within 80 kilometers (50 miles) of ANL-W in the year 2010, and a maximally exposed offsite individual (a member of the public assumed to be residing at the INEEL site boundary and receiving the maximum dose). As explained in Appendix E, Section E.4.1, the dose resulting from the release of tritium depends heavily on the chemical form. The inhalation dose from oxidized tritium (HTO or T_2O) is 25,000 times higher than for elemental tritium (HT or T_2). In the environment, about 1 percent of elemental tritium would be oxidized over the long term. In this analysis, about 1 percent of the tritium conservatively was assumed to be in oxidized form at the time of release.

Table 4–12 Annual and Project Total Radiological Impacts to the Public From Operational Activities Under Alternative 1

Receptor	*Electrometallurgically Treat Driver Spent Nuclear Fuel*	*Electrometallurgically Treat Blanket Spent Nuclear Fuel*	*Total*
Population Within 80 Kilometers (50 Miles) in the Year 2010			
Collective dose (person-rem per year) [a]	0.0027	0.000083	0.0028
Excess latent cancer fatalities (per year)	1.4×10^{-6}	4.2×10^{-8}	1.4×10^{-6}
Project total excess latent cancer fatalities [b]	8.0×10^{-6}	2.2×10^{-7}	8.2×10^{-6}
Maximally Exposed Offsite Individual			
Dose (millirem per year) [a]	0.00033	0.000010	0.00034
Percent of annual background radiation [c]	0.000092	2.8×10^{-6}	0.000094
Latent cancer fatality risk (per year)	1.6×10^{-10}	5.0×10^{-12}	1.7×10^{-10}
Project total lifetime cancer fatality risk [b]	9.6×10^{-10}	2.6×10^{-11}	9.9×10^{-10}
Average Individual Within 80 Kilometers (50 Miles)			
Dose (millirem per year) [d]	0.000011	3.5×10^{-7}	0.000012
Latent cancer fatality risk (per year)	5.6×10^{-12}	1.7×10^{-13}	5.8×10^{-12}
Project total lifetime cancer fatality risk [b]	3.3×10^{-11}	9.1×10^{-13}	3.4×10^{-11}

[a] Annual maximum dose during normal operations.
[b] Total calculated risk over 13 years of emissions.
[c] The annual natural background radiation level at INEEL is about 360 millirem for the average individual (see Table 3–8); the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 person-rem.
[d] Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of ANL-W in the year 2010 (240,338).

Primary contributors to doses to members of the public are from releases of tritium gas and krypton-85, which together contribute over 99.9 percent of the total calculated doses. To put the operational impacts into perspective, comparisons with impacts from natural background radiation also are included in Table 4–12. As shown in the table, the expected radiation doses to the maximally exposed offsite individual and the general public are much smaller than the limit of 10 millirem per year set by the EPA (40 CFR 61) and DOE (DOE Order 5400.5).

Occupational doses were estimated by examining the type and duration of various operations performed by ANL-W workers involved with the electrometallurgical treatment of sodium-bonded spent nuclear fuel. The estimated annual worker population collective dose would be 22 person-rem, with an average individual dose of 60 millirem per year for each of the 346 involved workers. If these estimates were extended over the 13 years of electrometallurgical treatment activities, and a 1-year dose (33 person-rem) from deactivation activities is included, the project total worker population dose would be 319 person-rem, leading to a risk of 0.13 latent cancer fatalities (see **Table 4–13**).

Table 4–13 Annual and Project Total Radiological Impacts to Workers From Operational Activities Under Alternative 1

Receptor	*Impacts*
Worker [a]	
Average worker dose (millirem per year)	60
Average worker latent cancer fatality risk project total over 13 years	0.00031
Worker Population	
Collective dose (person-rem per year)	22
Excess latent cancer fatalities (per year)	0.0088
Project total dose (person-rem)	319
Project total excess latent cancer fatalities	0.13

[a] The regulatory dose limit for an individual worker is 5,000 millirem per year (10 CFR 835). However, the maximum annual dose to a worker would be kept below the DOE Administrative Control Level of 2,000 millirem per year as established for all DOE activities in DOE Order N 441.1.

Source: ANL 1999.

As shown in Tables 4–12 and 4–13:

- The maximum dose to the maximally exposed offsite individual would be 0.00034 millirem per year, with an associated risk of developing a fatal cancer of 1.7×10^{-10} per year (or one chance in 5.9 billion that the individual would develop a fatal cancer per year of exposure).

- The maximum collective dose to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 0.0028 person-rem per year, with an associated 1.4×10^{-6} latent cancer fatalities per year (or one chance in 667,000 that the population would experience a fatal cancer per year of exposure).

- The project total risk to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 8.2×10^{-6} latent cancer fatalities (or one chance in 122,000 that the exposed population would experience a fatal cancer).

- The collective dose to facility workers would be 22 person-rem per year, with an associated 0.0088 latent cancer fatalities per year (or one chance in 113 that the workers would experience a fatal cancer per year of operation).

- The project total dose to facility workers would be 319 person-rem, with an associated 0.13 latent cancer fatalities (or one chance in seven that the workers would experience a fatal cancer).

These results indicate that no increase in the expected number of fatal cancers among the various affected populations would result from the activities under this alternative.

Hazardous Chemical Impacts

The hazardous chemical impacts to the public residing at the INEEL site boundary from the operational activities at ANL-W under this alternative are summarized in **Table 4–14**. Appendix E, Section E.5, provides details on the model used and results obtained. The results, presented in Table 4–14, indicate that no adverse toxic health or cancer effects would be expected from exposure to hazardous chemicals released under this alternative. The existing chemical environment is presented in Section 3.2.10.2.

Table 4–14 Hazardous Chemical Impacts to the Public From Operational Activities Under Alternative 1

Chemical	*Maximum Annual Concentration (milligrams per cubic meter)*	*Hazard Quotient (noncarcinogenic chemicals)*	*Cancer Risk (carcinogenic chemicals)*
1,3-Butadiene	3.6×10^{-8}	None	9.9×10^{-9}
Acetaldehyde	2.3×10^{-8}	2.5×10^{-6}	5.0×10^{-11}
Acrolein	7.1×10^{-9}	0.00035	None
Benzene	6.9×10^{-7}	None	5.4×10^{-9}
Cadmium	3.6×10^{-13}	None	6.5×10^{-13}
Formaldehyde	7.1×10^{-8}	None	9.2×10^{-10}
Toluene	2.5×10^{-7}	6.2×10^{-7}	None
Hazard Index		0.00036	Not applicable

4.3.4.2 Facility Accidents

Radiological Impacts

Potential radiological impacts to the public and a noninvolved onsite worker due to accidents during electrometallurgical treatment operational activities are summarized and presented in this section. Since electrometallurgical treatment processes are performed in both the Fuel Conditioning Facility and the Hot Fuel Examination Facility, accidents at both facilities would be considered. The detailed analysis of facility accidents, with the associated assumptions, is presented in Appendix F. The detailed analysis considered a wide spectrum of potential accident scenarios including fire, spills, criticality, earthquake, and aircraft crash. Aircraft crash and criticality accidents were determined to have an accident frequency of less than 10^{-7} per year, and were not analyzed further. **Table 4–15** presents the frequencies and consequences of the postulated set of accidents to the maximally exposed offsite individual; the offsite population residing within 80 kilometers (50 miles) of the facility; and a noninvolved worker located 100 meters (330 feet) to 230 meters (755 feet) from the facility. The 230-meter (755-foot) distance is the ANL-W bus staging area, which leads to a higher dose to the noninvolved worker for the scenarios with elevated releases.

The dose to the maximally exposed offsite individual was calculated for 95th percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50th percentile meteorological conditions. DOE did not quantitatively estimate the involved worker dose due to accidents (see discussions on the involved worker in Section 4.2.4.2). The accident risks are summarized in **Table 4–16**.

Table 4–15 Accident Frequency and Consequences Under Alternative 1

Accident	Frequency (event per year)	Maximally Exposed Offsite Individual		Population Within 80 Kilometers (50 Miles)		Noninvolved Worker	
		Dose (millirem)	Latent Cancer Fatality Risk [a]	Dose (person-rem)	Excess Latent Cancer Fatalities [b]	Dose (millirem)	Latent Cancer Fatality Risk [a]
Driver Spent Nuclear Fuel							
Salt powder spill	0.01	0.00046	2.3×10^{-10}	0.000098	4.9×10^{-8}	4.7×10^{-7}	1.9×10^{-13}
Salt transfer drop	1.0×10^{-7}	0.19	9.5×10^{-8}	0.022	0.000011	0.073	2.9×10^{-8}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Cask drop	0.01	0.030	1.5×10^{-8}	0.0035	1.8×10^{-6}	0.00084	3.4×10^{-10}
Design-basis earthquake	0.008	12	6.0×10^{-6}	1.4	0.0007	4.7	1.9×10^{-6}
Beyond-design-basis earthquake	0.00001	22,000	0.022	2,500	1.3	370	0.00015
Blanket Spent Nuclear Fuel							
Salt powder spill	0.01	0.00015	7.5×10^{-11}	0.000033	1.7×10^{-8}	1.3×10^{-6}	5.3×10^{-13}
Salt transfer drop	1.0×10^{-7}	0.065	3.3×10^{-8}	0.0077	3.9×10^{-6}	0.22	8.8×10^{-8}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Cask drop	0.01	0.0024	1.2×10^{-9}	0.00028	1.4×10^{-7}	0.000049	2.0×10^{-11}
Design-basis earthquake	0.008	4.0	2.0×10^{-6}	0.47	0.00024	14	5.6×10^{-6}
Beyond-design-basis earthquake	0.00001	930	0.00047	110	0.055	560	0.00023

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

Table 4–16 Annual Cancer Risks Due to Accidents Under Alternative 1

Accident	Maximally Exposed Offsite Individual [a]	Population Within 80 Kilometers (50 Miles) [b]	Noninvolved Worker [a]
Driver Spent Nuclear Fuel			
Salt powder spill	2.3×10^{-12}	4.9×10^{-10}	1.9×10^{-15}
Salt transfer drop	9.5×10^{-15}	1.1×10^{-12}	2.9×10^{-15}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Cask drop	1.5×10^{-10}	1.7×10^{-8}	3.4×10^{-12}
Design-basis earthquake	4.8×10^{-8}	5.6×10^{-6}	1.5×10^{-8}
Beyond-design-basis earthquake	2.2×10^{-7}	0.000013	1.5×10^{-9}
Blanket Spent Nuclear Fuel			
Salt powder spill	7.5×10^{-13}	1.7×10^{-10}	5.3×10^{-15}
Salt transfer drop	3.3×10^{-15}	3.9×10^{-13}	8.8×10^{-15}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Cask drop	1.2×10^{-11}	1.4×10^{-9}	2.0×10^{-13}
Design-basis earthquake	1.6×10^{-8}	1.9×10^{-6}	4.5×10^{-8}
Beyond-design-basis earthquake	4.7×10^{-9}	5.5×10^{-7}	2.3×10^{-9}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

For accidents at ANL-W, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be 2.2×10^{-7} per year (or one chance in 4.5 million that the individual would develop a fatal cancer per year of operation) and 4.5×10^{-8} per year (or one chance in 22.2 million that the individual would develop a fatal cancer per year of operation), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.000013 per year (or one chance in 76,920 that the population would experience a fatal cancer per year of operation).

Hazardous Chemical Impacts

Nonradiological impacts are evaluated in terms of comparison to ERPGs. ERPG values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects (see Appendix F, Section F.3.1.2, for details). The nonradiological impacts of potential facility accidents associated with the electrometallurgical treatment alternative at ANL-W are summarized in **Table 4–17**.

Table 4–17 Hazardous Chemical Accident Impacts Under Alternative 1

Accident	*Frequency (event per year)*	*Receptor*	*Exposure*
Uranium handling accident	0.01	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Design-basis earthquake	0.0002	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Uranium fire	0.00001	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Beyond-design-basis earthquake	0.00001	Noninvolved worker	Cadmium: less than ERPG-1
			Uranium: less than ERPG-1
		Maximally exposed offsite individual	Cadmium: less than ERPG-1
			Uranium: less than ERPG-1

ERPG = Emergency Response Planning Guideline.

4.3.5 Environmental Justice

As discussed in Appendix H, Executive Order 12898 directs Federal agencies to address disproportionately high and adverse health or environmental effects of alternatives on minority or low-income populations.

Analyses of normal operations and accident conditions presented in Section 4.3.4 show the risk of latent cancer fatalities to the public residing within 80 kilometers (50 miles) of the electrometallurgical processing facilities at ANL-W to be much lower than 1. Therefore, radiological and nonradiological risks posed by implementation of this alternative would have no disproportionately high and adverse consequences on any particular group within the general population, including minority or low-income populations.

4.3.6 Waste Management

Electrometallurgical treatment of sodium-bonded spent nuclear fuel at ANL-W would generate process waste from treatment operations, other associated process waste from normal support operations, and deactivation waste following the conclusion of operations. Process waste would include fuel hardware and high-level radioactive metallic and ceramic waste. Other associated process waste would include operational waste such as failed equipment, rags, packaging materials, and other miscellaneous items. Deactivation waste would include the disposal of process equipment and other materials. All of these materials would be categorized according to existing DOE Orders and ANL-W waste management procedures. The anticipated categorization

of waste types and their expected interim storage and final disposal locations are given in Table 4–9 (see Section 4.2.6). The quantities of various waste forms generated as a result of electrometallurgical treatment at ANL-W are provided in **Table 4–18.**

Table 4–18 Amounts of Waste Generated Under Alternative 1[a]

Waste Stream	*Waste Quantities*	
	Volume (cubic meters)	*Mass (kilograms)*
Direct Process Waste		
Fuel assembly hardware (low-level radioactive waste)	12.5	6,600
High-level radioactive ceramic waste	78 (125 canisters)[b]	120,000
High-level radioactive metallic waste	3.1 (5 canisters)[b]	9,000
Other Associated Process Waste		
Low-level radioactive waste[c]	706	143,000
Transuranic waste	12.5	5,400
Mixed waste	35.3	19,000
Sanitary waste	4,960	1.72×10^6
Deactivation Waste		
Low-level radioactive waste[c]	143	48,000
Transuranic waste	1.6	853
Mixed waste	4.2	2,900

[a] These waste generation estimates are through the year 2015. This is the assumed date that these materials might be sent to the repository. Treatment, high-level radioactive waste processing, deactivation, and interim storage would be accomplished during this time period.

[b] Standardized canisters.

[c] The volumes listed represent final disposal volumes following volume reduction at the Waste Experimental Reduction Facility at INEEL.

Source: ANL 1999.

Estimates of the total amount of other associated process waste that would be generated are based on an evaluation of waste forecasts from ANL-W that account only for the fraction of total ANL-W waste that would be attributable to the processing of sodium-bonded spent nuclear fuel under this alternative. The values in Table 4–18 are for disposal and account for volume reduction. It is anticipated that a large fraction of the low-level radioactive waste generated as a result of electrometallurgical treatment could be volume-reduced at the Waste Experimental Reduction Facility at INEEL prior to disposal at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex.

The waste values in Table 4–18 are total quantities that would be generated as a result of Alternative 1 operations. They are not incremental increases over the volumes provided in Table 4–10 that would result from the No Action Alternative. In Alternative 1, the sodium-bonded spent nuclear fuel would be transformed into high-level radioactive waste forms (ceramic and metallic) for disposal in the repository, and in this conversion process, the total volume of material to be disposed of in the repository would be reduced from direct disposal values.

Direct Process Waste

For electrometallurgical treatment, fuel assembly hardware would be removed from the fuel elements in the Fuel Conditioning Facility air cell and disposed of as low-level radioactive waste. These components are primarily stainless steel materials that contain short-lived radionuclides. This waste stream has been produced

at ANL-W for many years and would be handled, as in the past, according to DOE Orders and ANL-W waste management procedures.

Under Alternative 1, metallic and ceramic high-level radioactive waste would be a primary product. The salt removed from the electrorefiners would contain the majority of fission products and transuranics from the spent nuclear fuel. This removed salt would be packaged and transferred to the Hot Fuel Examination Facility for processing into ceramic waste. The metallic waste form would consist primarily of stainless steel cladding hulls containing the noble metal fission products. The hulls would be removed from the electrorefiner and packaged for shipment to the Hot Fuel Examination Facility for processing into the metallic waste form. Both the ceramic and metallic waste would be categorized as high-level radioactive waste. The volumes of waste forms provided in Table 4–18 are for the standardized canisters required for disposal of these materials.

The metallic and ceramic high-level radioactive waste generated would be stored temporarily at the Radioactive Scrap and Waste Facility at ANL-W to allow retrieval for future disposal. The Radioactive Scrap and Waste Facility was designed and constructed for temporary storage of this type of waste. Shielding would be provided by a combination of (1) steel storage liners which would store the waste, and (2) the soil surrounding the liners. When a geologic repository is available, the waste cans containing the metallic and ceramic high-level radioactive waste would be removed from storage, shipped to the INEEL Dry Transfer Facility, and prepared for shipment to the repository.

Other Associated Process Low-Level Radioactive Waste

Low-level radioactive waste would be generated as a result of electrometallurgical treatment of sodium-bonded spent nuclear fuel at ANL-W. This would result from activities in the Fuel Conditioning Facility and the Hot Fuel Examination Facility (e.g., equipment decontamination and repair), as well as in other facilities at ANL-W (e.g., analytical laboratory activities). Material in this waste stream has been generated and routinely handled at ANL-W for many years.

The volume of low-level radioactive waste resulting from electrometallurgical treatment at ANL-W that would require disposal (after volume reduction) would be approximately 48 cubic meters (1,695 cubic feet) per year. This represents approximately 0.08 percent of the total annual volume of low-level radioactive waste currently being disposed of at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex, and the total of 706 cubic meters (24,932 cubic feet) represents approximately 0.9 percent of the total Radioactive Waste Management Complex disposal capacity.

Other Associated Process Transuranic Waste

Transuranic waste would be generated by decontamination activities for repair and maintenance of items, and miscellaneous work associated with the electrometallurgical processing. Transuranic waste would be generated primarily from activities conducted in gloveboxes and hot cells at ANL-W.

All of the transuranic waste generated would be packaged and certified in accordance with Waste Isolation Pilot Plant Acceptance Criteria prior to transport to the Waste Isolation Pilot Plant. The transuranic waste generated would amount to approximately 1 cubic meter (35 cubic feet) per year, which is less than 0.002 percent of the volume of transuranic waste in retrievable storage at the Radioactive Waste Management Complex at INEEL. The total volume of transuranic waste is 12.5 cubic meters (441 cubic feet), which is less than 0.008 percent of the estimated total volume of transuranic waste to be placed at the Waste Isolation Pilot Plant.

Other Associated Process Mixed Waste

Mixed waste of this category would be generated primarily from the disposal of any cadmium-contaminated equipment or cleanup material and the analysis of cadmium samples. Mixed waste would be handled according to ANL-W procedures that require limited accumulation at the point of generation. Interim storage of this waste would be accomplished at the Radioactive Scrap and Waste Facility prior to eventual disposal. The Radioactive Scrap and Waste Facility is a permitted mixed waste storage facility for these materials. The mixed waste streams that contribute to the overall mixed waste generated by electrometallurgical treatment have been identified in the INEEL Site Treatment Plan (DOE 1995b).

Deactivation Waste

A variety of waste would be generated as part of deactivation activities associated with electrometallurgical treatment processing at ANL-W. This would include process equipment and process material, such as electrorefiner cadmium. Generated waste categories would include low-level radioactive waste, transuranic waste, and mixed waste. This waste would be categorized and disposed of according to DOE Orders and ANL-W radioactive waste management procedures, as described above for each waste category.

The largest volume of deactivation waste would be low-level radioactive waste, transuranic waste, and mixed waste generated as a result of equipment dismantling and disposal. Components that would require disposition include two electrorefiners, two hot isostatic presses, and two V-mixers, as well as other components such as the grinder/crusher. Decontamination of these components would generate additional mixed, transuranic, and low-level radioactive waste that would require management. The deactivation waste volume would be generated in a single year. This waste would represent an increase of approximately 3.5 times the annual waste generated by electrometallurgical treatment requiring disposal. The total deactivation waste would represent approximately 20 percent over the total associated process waste (excluding sanitary waste) requiring disposal.

4.4 Alternative 2: Clean and Package Blanket Fuel in High-Integrity Cans and Electrometallurgically Treat Driver Fuel at ANL-W

Under this alternative, the sodium-bonded blanket spent nuclear fuel would be cleaned to remove metallic sodium and placed in high-integrity cans. These cans then would be placed into overpack containers prior to dry storage at the Radioactive Scrap and Waste Facility, pending repackaging and transportation for disposal in a geologic repository. The removed sodium contains radioactive elements, principally cesium. The cesium would be separated from the sodium and stabilized as ceramic waste. The sodium would be stabilized using an oxidation/carbonation process (see Appendix C for more detail) (ANL 1999). The sodium-bonded driver spent nuclear fuel would be treated at ANL-W using the electrometallurgical treatment process. The process steps for the electrometallurgical treatment of driver spent nuclear fuel would be similar to those described earlier in Section 4.3 and in Appendix C. The treatment of driver spent nuclear fuel using the electrometallurgical process could start as early as 2000 and could be completed by 2006 to 2007. The preparation of blanket spent nuclear fuel and its placement in high-integrity cans could start in 2003 and could be completed by 2009. In addition, a full year of operations would be needed to deactivate the processing equipment and establish an industrially safe configuration.

4.4.1 Air Quality

Nonradiological Gaseous Emissions

It is expected that criteria and hazardous air pollutants released from operational activities at ANL-W under this alternative would be the same as for Alternative 1, as described in Section 4.3.1 (see also Appendix E, Section E.5.3.1, for more detail).

Radiological Gaseous Emissions

The cleaning of the blanket spent nuclear fuel to remove metallic sodium and the electrometallurgical treatment of the driver spent nuclear fuel would release gaseous fission products to the argon cell environment. Krypton-85 and elemental tritium are the most prevalent radioactive gaseous fission products that would be released to the environment. The released tritium in the cell would not be oxidized due to a very low presence of oxygen and humidity in the argon cell. Appendix E, Section E.4.2, provides details on releases during the processing period at ANL-W. The argon cell also contains an equilibrium concentration of other radionuclide isotopes. Appendix E, Section E.4.1, provides a list of various isotopes that are present in the argon cell in nanocuries (10^{-9} curies) and are released to the atmosphere through the facility stack, along with krypton-85 and elemental tritium. The maximum release of radioactive gaseous emissions would occur when cleaning blanket spent nuclear fuel for placement in high-integrity cans and electrometallurgical treatment of driver spent nuclear fuel are performed simultaneously. This simultaneous operation was estimated to occur over a three-year period starting in 2003. Based on an annual cleaning throughput of 10 metric tons of heavy metal of blanket spent nuclear fuel and an electrometallurgical treatment processing of about 0.6 metric tons of heavy metal of driver spent nuclear fuel, about 809 curies of elemental tritium and 11,860 curies of gaseous krypton-85 would be released annually to the atmosphere. The radiological exposures to the public and workers from radioactive emissions are presented in Section 4.4.4.

4.4.2 Water Resources

Surface Water

No surface water is used at ANL-W. Flood waters from the Big Lost River would not be expected to reach the facilities at ANL-W, as shown in Figure 3–3.

Nonradiological Liquid Effluent

There are no discharges to the surface waters at ANL-W, except for discharges of nonhazardous liquid waste to the sewage pond and to the industrial waste pond. Big Lost River, Little Lost River, and Birch Creek would not be impacted by activities associated with high-integrity can operations and electrometallurgical treatment process operations. Current operating and monitoring practices would continue for NPDES stormwater and liquid effluent discharges associated with facilities at ANL-W (see also Section 4.2.2).

During fuel treatment and associated activities, some hazardous materials may be used inside buildings. To prevent potential releases to surface or subsurface waters resulting from spills of hazardous materials used in buildings, facilities are designed, constructed, and maintained to contain these materials. Double-contained pipes, leak detection, and secondary containment of tanks are some of the features used to prevent hazardous material releases to the environment. Following existing written procedures, spill containment and cleanup equipment is present in areas where hazardous materials are stored or used (DOE 1996b).

Radiological Liquid Effluent

No radiological liquid effluent or waste generated by high-integrity can and electrometallurgical treatment process operations would be discharged to surface water.

Groundwater

Under this alternative at ANL-W, there would be little change in groundwater consumption for domestic use since there is little change expected in the number of workers. The current water usage at ANL-W is 188 million liters (49.6 million gallons) per year.

Nonradiological Liquid Effluent

No nonradiological liquid effluent generated by high-integrity can and electrometallurgical treatment process operations would be discharged to groundwater.

Radiological Liquid Effluent

No radiological liquid effluent or waste generated by high-integrity can and electrometallurgical treatment process operations would be discharged to groundwater.

4.4.3 Socioeconomics

Under this alternative, the existing facilities at ANL-W would remain operational. No new employment or in-migration of workers would be required. Thus, there would be no additional impacts on the socioeconomic conditions in the region around ANL-W and INEEL.

4.4.4 Public and Occupational Health and Safety

The assessments of potential incremental radiological and chemical impacts associated with this alternative are presented in this section. Summaries of radiological impacts from normal operations are presented in Tables 4–19 and 4–20 for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in Tables 4–21 and 4–22. The impacts from hazardous chemical releases during accident conditions are presented in Table 4–23. Background information on the effects of radiation on human health and safety is presented in Appendix E, Section E.2.

4.4.4.1 Normal Operations

Radiological Impacts

Under this alternative, radioactive releases would occur during sodium-bonded blanket spent nuclear fuel cleaning and driver spent nuclear fuel chopping. All of these activities would be performed in the argon cell. Appendix E, Sections E.3, E.4.1, and E.4.2, details the method and assumptions used for calculating the impacts of normal operational radiological releases on the public health and safety. The maximum annual dose to the public would result when cleaning of blanket spent nuclear fuel and treatment of driver spent nuclear fuel are performed simultaneously under this alternative. Appendix E, Section E.4.2, provides details on the treatment process duration and throughputs for each fuel type. The duration of the treatment process is estimated to be nine years.

Calculated maximum annual and project total radiological impacts to the public are given in **Table 4–19**. The impacts are calculated for two types of receptors: the general public living within 80 kilometers (50 miles) of ANL-W in 2010, and a maximally exposed offsite individual (a member of the public assumed to be residing at the INEEL site boundary and receiving the maximum dose). Primary contributors to doses to members of the public are releases of tritium gas (about 1 percent of which was conservatively assumed to be in oxidized form) and krypton-85, which together contribute over 99.9 percent of the total calculated doses. To put the operational impacts into perspective, comparisons with impacts from natural background radiation also are included in the table. As shown in this table, the expected radiation doses to the maximally exposed offsite individual and the general public are much smaller than the limit of 10 millirem per year set by the EPA (40 CFR 61) and DOE (DOE Order 5400.5).

Table 4–19 Annual and Project Total Radiological Impacts to the Public From Operational Activities Under Alternative 2

Receptor	*Electrometallurgically Treat Driver Spent Nuclear Fuel*	*Clean and Place Blanket Spent Nuclear Fuel in High-Integrity Cans*	*Total*
Population Within 80 Kilometers (50 Miles) in the Year 2010			
Collective dose (person-rem per year) [a]	0.0027	0.00028	0.0030
Excess latent cancer fatalities (per year)	1.4×10^{-6}	1.4×10^{-7}	1.5×10^{-6}
Project total excess latent cancer fatalities [b]	8.0×10^{-6}	3.4×10^{-7}	8.3×10^{-6}
Maximally Exposed Offsite Individual			
Dose (millirem per year) [a]	0.00033	0.000048	0.00038
Percent of annual background [c]	0.000092	0.000013	0.00011
Latent cancer fatality risk (per year)	1.7×10^{-10}	2.4×10^{-11}	1.9×10^{-10}
Project total lifetime cancer fatality risk	9.4×10^{-10}	5.8×10^{-11}	1.0×10^{-9}
Average Individual Within 80 Kilometers (50 Miles)			
Dose (millirem per year) [d]	0.000011	1.2×10^{-6}	0.000012
Latent cancer fatality risk (per year)	5.6×10^{-12}	5.8×10^{-13}	6.2×10^{-12}
Project total lifetime cancer fatality risk [b]	3.3×10^{-11}	1.4×10^{-12}	3.5×10^{-11}

[a] Annual maximum dose during normal operations.
[b] Total calculated risk over nine years.
[c] The annual natural background radiation level at INEEL is about 360 millirem for the average individual (see Table 3–8); the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 person-rem.
[d] Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of ANL-W in the year 2010 (240,338).

Table 4–20 summarizes worker population doses. Occupational doses were estimated by examining the type and duration of various operations performed by ANL-W workers involved with sodium-bonded spent nuclear fuel high-integrity can and electrometallurgical treatment processes. It was concluded that the average worker dose would not be different from that currently being experienced. The estimated annual collective worker dose would be 22 person-rem, with an average individual dose of 60 millirem per year for each of the 346 involved workers. If these estimates were extended over the nine years of treatment activities, plus one year for deactivation of the facilities, the project total worker population dose would be 231 person-rem, leading to a risk of 0.092 latent cancer fatalities.

As shown in Tables 4–19 and 4–20:

- The annual dose to the maximally exposed offsite individual would be 0.00038 millirem per year, with an associated 1.9×10^{-10} risk per year of developing a fatal cancer (or one chance in 5.3 billion that the individual would develop a fatal cancer per year of exposure).

- The collective dose to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 0.0030 person-rem per year, with an associated 1.5×10^{-6} latent cancer fatalities per year (or one chance in 666,700 that the population would experience a fatal cancer per year of exposure).

- The project total risk to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 8.3×10^{-6} latent cancer fatalities (or one chance in 120,000 that the exposed population would experience a fatal cancer).

Table 4–20 Annual and Project Total Radiological Impacts to Workers From Operational Activities Under Alternative 2

Receptor	*Impacts*
Worker [a]	
Average worker dose (millirem per year)	60
Average worker latent cancer fatality risk (project total over nine years)	0.00022
Worker Population	
Collective dose (person-rem per year) [b]	22
Excess latent cancer fatalities (per year)	0.0088
Project total dose (person-rem)	231
Project total excess latent cancer fatalities	0.092

[a] The regulatory dose limit for an individual worker is 5,000 millirem per year (10 CFR 835). However, the maximum annual dose to a worker would be kept below the DOE Administrative Control Level of 2,000 millirem per year, as established for all DOE activities in DOE Order N 441.1.

[b] Worker dose is 33 person-rem for one year of deactivation activities.

Source: ANL 1999.

- The collective dose to facility workers would be 22 person-rem per year, with an associated 0.0088 latent cancer fatalities per year (or one chance in 113 that the workers would experience a fatal cancer per year of operation).

- The project total dose to facility workers would be 231 person rem, with an associated 0.092 latent cancer fatalities (or one chance in 11 that the exposed workers would experience a fatal cancer).

These results indicate that no increase in the expected number of fatal cancers among the various affected populations would result from the activities under this alternative.

Hazardous Chemical Impacts

The hazardous chemical impacts to the public residing at the INEEL site boundary from the operational activities at ANL-W under Alternative 2 would be similar to the impacts evaluated for Alternative 1 described in Section 4.3.4.1. The results indicate that no adverse toxic health or cancer effects would be expected from exposure to hazardous chemicals released under this alternative. The existing chemical environment is described in Section 3.2.10.2.

4.4.4.2 Facility Accidents

Radiological Impacts

Potential radiological impacts to the public and a noninvolved onsite worker due to accidents during cleaning activities for placement of blanket spent nuclear fuel elements in high-integrity cans and the electrometallurgical treatment operational activities for driver spent nuclear fuel are summarized and presented in this section. The detailed analysis of facility accidents, with the associated assumptions, is presented in Appendix F.

The detailed analysis considered a wide spectrum of potential accident scenarios including fire, spills, criticality, earthquake, and aircraft crash. The aircraft crash and criticality events were determined to have an occurrence frequency of less than 10^{-7} per year, and consequence analyses for these two events were not

performed. Cleaning of the blanket spent nuclear fuel would be performed in the Hot Fuel Examination Facility; treatment of the driver spent nuclear fuel would be performed in both the Hot Fuel Examination Facility and the Fuel Conditioning Facility. Because driver spent nuclear fuel processing would take place in both of these facilities, the beyond-design-basis earthquake event is assessed for the driver spent nuclear fuel, taking into account the multifacility impacts of this event. The cleaning of the blanket spent nuclear fuel would be performed only in the Hot Fuel Examination Facility. **Table 4–21** presents the frequencies and consequences of the postulated set of accidents to the maximally exposed offsite individual, the offsite population residing within 80 kilometers (50 miles) of the facility, and a noninvolved worker located 100 meters (330 feet) to 230 meters (755 feet) from the facility.

Table 4–21 Accident Frequency and Consequences at ANL-W Under Alternative 2

Accident	Frequency (event per year)	Maximally Exposed Offsite Individual		Population Within 80 Kilometers (50 Miles)		Noninvolved Worker	
		Dose (millirem)	Latent Cancer Fatality Risk [a]	Dose (person-rem)	Excess Latent Cancer Fatalities [b]	Dose (millirem)	Latent Cancer Fatality Risk [a]
Driver Spent Nuclear Fuel							
Salt powder spill	0.01	0.00046	2.3×10^{-10}	0.000098	4.9×10^{-8}	4.7×10^{-7}	1.9×10^{-13}
Salt transfer drop	1.0×10^{-7}	0.19	9.5×10^{-8}	0.022	0.000011	0.073	2.9×10^{-8}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Cask drop	0.01	0.030	1.5×10^{-8}	0.0035	1.8×10^{-6}	0.00084	3.4×10^{-10}
Design-basis earthquake	0.008	12	6.0×10^{-6}	1.4	0.0007	4.7	1.9×10^{-6}
Beyond-design-basis earthquake	0.00001	22,000	0.022	2,500	1.3	370	0.00015
Blanket Spent Nuclear Fuel							
Cask drop	0.01	0.0024	1.2×10^{-9}	0.00028	1.4×10^{-7}	0.000049	2.0×10^{-11}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Sodium fire [c]	0.008	5.9	3.0×10^{-6}	0.69	0.00035	0.054	2.2×10^{-8}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.
[c] The frequency for this accident is the frequency for the facility design-basis earthquake initiating cell fire.

The dose to the maximally exposed offsite individual was calculated for the 95th percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50th percentile meteorological conditions. DOE did not quantitatively estimate the involved worker dose due to accidents (see the discussion on the involved worker in Section 4.2.4.2). The accident risks for the same receptors are summarized in **Table 4–22**.

For accidents at ANL-W, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be 2.2×10^{-7} per year (or one chance in 4.5 million that the individual would develop a fatal cancer per year of operation) and 1.5×10^{-8} per year (or one chance in 66.7 million that the individual would develop a fatal cancer per year of operation), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.000013 per year (or one chance in 76,920 that the population would develop a fatal cancer per year of operation).

Table 4–22 Annual Cancer Risks Due to Accidents at ANL-W Under Alternative 2

Accident	*Maximally Exposed Offsite Individual* [a]	*Population Within 80 Kilometers (50 Miles)* [b]	*Noninvolved Worker* [a]
Driver Spent Nuclear Fuel			
Salt powder spill	2.3×10^{-12}	4.9×10^{-10}	1.9×10^{-15}
Salt transfer drop	9.5×10^{-15}	1.1×10^{-12}	2.9×10^{-15}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Cask drop	1.5×10^{-10}	1.7×10^{-8}	3.4×10^{-12}
Design-basis earthquake	4.8×10^{-8}	5.6×10^{-6}	1.5×10^{-8}
Beyond-design-basis earthquake	2.2×10^{-7}	0.000013	1.5×10^{-9}
Blanket Spent Nuclear Fuel			
Cask drop	1.2×10^{-11}	1.4×10^{-9}	2.0×10^{-13}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Sodium fire	2.4×10^{-8}	2.8×10^{-6}	1.7×10^{-10}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

Hazardous Chemical Impacts

Nonradiological impacts are evaluated in terms of comparison to ERPG. ERPG values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects (see Appendix F, Section F.3.1.2, for details).

The hazardous chemical impacts of potential facility accidents associated with the treatment of driver spent nuclear fuel using the electrometallurgical treatment process are summarized in **Table 4–23**.

Table 4–23 Hazardous Chemical Impacts Due to Accidents at ANL-W Under Alternative 2

Accident	*Frequency (event per year)*	*Receptor*	*Exposure*
Uranium handling accident	0.01	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Design-basis earthquake	0.0002	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Sodium fire	0.008	Noninvolved worker	Sodium: less than ERPG-1
		Maximally exposed offsite individual	Sodium: less than ERPG-1
Uranium fire	0.00001	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Beyond-design-basis earthquake	0.00001	Noninvolved worker	Cadmium: less than ERPG-1
			Uranium: less than ERPG-1
		Maximally exposed offsite individual	Cadmium: less than ERPG-1
			Uranium: less than ERPG-1

ERPG = Emergency Response Planning Guideline.

4.4.5 Environmental Justice

As discussed in Appendix H, Executive Order 12898 directs Federal agencies to address disproportionately high and adverse health or environmental effects of alternatives on minority or low-income populations.

Analyses of normal operations and accident conditions presented in Section 4.4.4 show the risk of latent cancer fatalities to the public residing within 80 kilometers (50 miles) of the electrometallurgical treatment processing facilities at ANL-W to be much lower than 1. Therefore, radiological and nonradiological risks posed by implementation of this alternative would have no disproportionately high and adverse consequences on any particular group within the general population, including minority or low-income populations.

4.4.6 Waste Management

This alternative would generate process waste from treatment operations, other associated process waste from normal support operations, and deactivation waste following the conclusion of operations. Process waste would include fuel assembly hardware and high-level radioactive metallic and ceramic waste. Other associated process waste would include operational waste such as failed equipment, rags, packaging materials, and other miscellaneous items. Deactivation waste would include the disposal of process equipment and other materials. All of these materials would be categorized according to existing DOE Orders and ANL-W waste management procedures. The anticipated categorization of these waste types and their expected interim storage and final disposal locations are given in Table 4–9 (see Section 4.2.6). The quantities of various waste forms generated as a result of Alternative 2 are provided in **Table 4–24**.

Table 4–24 Amounts of Waste Generated Under Alternative 2[a]

Waste Stream	*Waste Quantities*	
	Volume (cubic meters)	*Mass (kilograms)*
Direct Process Waste		
Fuel assembly hardware (low-level radioactive waste)	12.5	6,000
High-level radioactive ceramic waste	16.3 (26 canisters) [b]	24,400
High-level radioactive metallic waste	1.3 (2 canisters) [b]	2,500
Spent nuclear fuel	25.2 (63 canisters) [b]	63,000
Other Associated Process Waste		
High-level radioactive waste	0.4 (1 canister) [b]	220
Low-level radioactive waste [c]	555	113,000
Transuranic waste	9.1	3,800
Mixed waste	27.5	14,800
Sanitary waste	4,960	1.72×10^6
Deactivation Waste		
Low-level radioactive waste [c]	166.2	56,000
Transuranic waste	1.6	853
Mixed waste	4.8	3,200

[a] These waste generation estimates are through the year 2015. This is the assumed date that these materials might be sent to the repository. Treatment, high-level radioactive waste processing, deactivation, and interim storage would be accomplished during this time period.

[b] Standardized canisters.

[c] The volumes listed represent final disposal volumes following volume reduction at the Waste Experimental Reduction Facility at INEEL.

Source: ANL 1999.

Estimates of the total amount of other associated process waste that would be generated are based on an evaluation of waste forecasts from ANL-W that accounted only for the fraction of total ANL-W waste that would be attributable to the treatment of sodium-bonded spent nuclear fuel. The values in Table 4–24 are for disposal and account for volume reduction. It is anticipated that a large fraction of the low-level radioactive waste that would be generated as a result of Alternative 2 could be volume-reduced at the Waste Experimental Reduction Facility at INEEL prior to disposal at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex.

The waste values in Table 4–24 are total quantities that would be generated as a result of Alternative 2 operations. They are not incremental increases over the volumes provided in Table 4–10 that would result from the No Action Alternative. In Alternative 2, the driver spent nuclear fuel would be transformed into high-level radioactive waste forms (ceramic and metallic) for disposal in the repository. The blanket spent nuclear fuel would be cleaned and packaged in high-integrity cans for disposal in the repository. In this conversion process, the total volume of material to be disposed of in the repository would be reduced from direct disposal values.

Direct Process Waste

For this alternative, fuel hardware would be removed from the fuel elements in the Fuel Conditioning Facility air cell and disposed of as low-level radioactive waste. These components are primarily stainless steel materials that contain short-lived radionuclides. This waste stream has been produced at ANL-W for many years and would be handled, as in the past, according to DOE Orders and ANL-W waste management procedures.

Under this alternative, metallic and ceramic high-level radioactive waste would be a primary product of the electrometallurgical treatment of driver spent nuclear fuel. The salt removed from electrorefiners would contain the majority of fission products and transuranics from the spent nuclear fuel. This removed salt would be packaged and transferred to the Hot Fuel Examination Facility for processing into ceramic waste. The metallic waste form would consist primarily of stainless steel cladding hulls containing the noble metal fission products. The hulls would be removed from the electrorefiner and packaged for shipment to the Hot Fuel Examination Facility for processing into the metallic waste form. Both the ceramic and metallic waste would be categorized as high-level radioactive waste.

The packaged spent nuclear fuel volume is based on placing the blanket spent nuclear fuel in high-integrity cans which would be placed in standardized canisters. The volumes of waste forms provided in Table 4–24 are for the standardized canisters required for disposal of these materials.

The metallic and ceramic high-level radioactive waste generated as a result of electrometallurgical treatment of driver spent nuclear fuel under this alternative, driver and blanket spent nuclear fuel generated during the demonstration project, and packaged spent nuclear fuel would be stored temporarily at the Radioactive Scrap and Waste Facility at ANL-W to allow retrieval for future disposal. The Radioactive Scrap and Waste Facility was designed and constructed for temporary storage of this type of waste. Shielding would be provided by a combination of (1) steel storage liners which would store the waste, and (2) the soil surrounding the liners. When a geologic repository is available, the waste cans containing these materials would be removed from storage, shipped to the INEEL Dry Transfer Facility, and prepared for shipment to the repository.

Other Associated Process High-Level Radioactive Waste

High-level radioactive waste could be generated as a result of blanket spent nuclear fuel processing at ANL-W. This would result from activities in the Hot Fuel Examination Facility. Material in this waste stream would

consist of the absorbent used in the off-gas system which has collected the volatile radionuclides released from the spent nuclear fuel when heated.

The volume of high-level radioactive waste generated is expected to be less than the amount needed to fill a single standardized waste canister. Conservatively, the volume of a single canister, 0.4 cubic meters, has been used for the volume of high-level radioactive waste generated.

Other Associated Process Low-Level Radioactive Waste

Low-level radioactive waste would be generated as a result of processing at ANL-W. This would result from activities in the Fuel Conditioning Facility and the Hot Fuel Examination Facility (e.g., equipment decontamination and repair), as well as in other facilities at ANL-W (e.g., analytical laboratory activities). Material in this waste stream has been generated and routinely handled at ANL-W for many years.

The volume of low-level radioactive waste at ANL-W that would require disposal (after volume reduction) would be approximately 50 cubic meters (1,766 cubic feet) per year. This represents approximately 1 percent of the total annual volume of low-level radioactive waste currently being disposed of at the INEEL Subsurface Disposal Area in the Radioactive Waste Management Complex, and the total of 555 cubic meters (19,600 cubic feet) represents approximately 0.5 percent of the total Radioactive Waste Management Complex disposal capacity.

Other Associated Process Transuranic Waste

Transuranic waste would be generated by Alternative 2 from decontamination activities for repair and maintenance of items, and miscellaneous work associated with processing the sodium-bonded spent nuclear fuel. Transuranic waste would be generated primarily from activities conducted in gloveboxes and hot cells at ANL-W.

All of the transuranic waste generated as a result of the treatment of sodium-bonded spent nuclear fuel at ANL-W would be packaged and certified in accordance with Waste Isolation Pilot Plant Acceptance Criteria prior to transport to the Waste Isolation Pilot Plant. The transuranic waste generated would amount to approximately 1 cubic meter (35 cubic feet) per year, which is less than 0.002 percent of the volume of transuranic waste in retrievable storage at the Radioactive Waste Management Complex at INEEL. The total volume of transuranic waste is 9.1 cubic meters (321 cubic feet), which is 0.005 percent of the estimated total volume of transuranic waste to be placed at the Waste Isolation Pilot Plant.

Other Associated Process Mixed Waste

Mixed waste would be generated primarily from the disposal of any cadmium-contaminated equipment or cleanup material and the analysis of cadmium samples. Mixed waste would be handled according to ANL-W procedures that require limited accumulation at the point of generation. Interim storage of this waste would be accomplished at the Radioactive Scrap and Waste Facility prior to eventual disposal. The Radioactive Scrap and Waste Facility is a permitted mixed waste storage facility for these materials. The mixed waste streams that contribute to the overall mixed waste generated by electrometallurgical treatment have been identified in the INEEL Site Treatment Plan (DOE 1995b).

Deactivation Waste

A variety of waste would be generated as part of deactivation activities associated with electrometallurgical treatment processing at ANL-W. This would include process equipment and process material, such as electrorefiner cadmium from electrometallurgical treatment of driver spent nuclear fuel. Generated waste

categories would include low-level radioactive waste, transuranic waste, and mixed waste. This waste would be categorized and disposed of according to DOE Orders and ANL-W radioactive waste management procedures, as described above for each waste category.

The largest volume of deactivation waste would be low-level radioactive waste generated as a result of equipment dismantling and disposal. Components of electrometallurgical treatment of the driver spent nuclear fuel that would require disposition include two electrorefiners, two hot isostatic presses, and two V-mixers, as well as other components such as the grinder/crusher. Decontamination of these components would generate additional mixed, transuranic, and low-level radioactive waste that would require management. The deactivation waste volume would be generated in two years. The total deactivation waste would represent an additional 30 percent over the total associated process waste (excluding sanitary waste) requiring disposal.

4.5 Alternative 3: Declad and Clean Blanket Fuel and Electrometallurgically Treat Driver Fuel at ANL-W; PUREX Process Blanket Fuel at SRS

Under this alternative, the sodium-bonded blanket spent nuclear fuel would be declad and cleaned to remove metallic sodium, packaged in aluminum cans at ANL-W, and shipped to SRS for treatment using the PUREX process at F-Canyon. The removed sodium would be stabilized using an oxidation/carbonation process (ANL 1999). The sodium-bonded driver spent nuclear fuel would be treated at ANL-W using the electrometallurgical treatment process. The high-level radioactive waste generated from the treatment of the blanket spent nuclear fuel at SRS would be in the form of borosilicate glass and would be stored at the SRS Defense Waste Processing Facility, pending repackaging and transportation for disposal in a geologic repository. The process steps for the electrometallurgical treatment of driver spent nuclear fuel would be similar to those described earlier in Section 4.3 and in Appendix C. The treatment of driver spent nuclear fuel using the electrometallurgical process could start as early as 2000 and could be completed by 2006 to 2007. The preparation of blanket spent nuclear fuel and its shipment to SRS could start in 2003 and could be completed by 2009. In addition, a full year of operations would be needed to deactivate the processing equipment and establish an industrially safe configuration.

PUREX processing of blanket spent nuclear fuel at SRS would require six months of operation and could be completed by 2010.

4.5.1 Air Quality

Nonradiological Gaseous Emissions

It is expected that criteria and hazardous air pollutants released from operational activities at ANL-W under this alternative to be the same as for Alternative 1, as described in Section 4.3.1, (see also Appendix E, Section E.5.3.1 for more detail). Baseline air quality concentrations are presented in Section 3.2.3.1.

The concentrations of nonradiological air pollutants attributed to this alternative at SRS are presented in **Table 4–25**. These concentrations are based on information in the *Savannah River Site Spent Nuclear Fuel Management Final Environmental Impact Statement* (SRS Spent Nuclear Fuel Management Final EIS) (DOE 2000) for the PUREX processing of similar fuel. See Appendix E, Section E.5.3.2, for more details. The site boundary concentrations are equal to the incremental concentrations generated in this alternative plus the baseline concentrations given in Section 3.3.3.1. Only those air pollutants that are expected and have ambient air quality standards are presented in the table. Note that there are no Prevention of Significant Deterioration increment-consuming sources at SRS; therefore, a Prevention of Significant Deterioration increment analysis was not performed.

Table 4–25 Nonradiological Air Quality Concentrations at the Site Boundary Under Alternative 3 at SRS for Comparison With Ambient Air Quality Standards

Pollutant	*Averaging Period*	*Most Stringent Standard or Guideline (micrograms per cubic meter)*	*Maximum Incremental Concentration (micrograms per cubic meter)*
Criteria Pollutants			
Carbon monoxide	8 hours 1 hour	10,000 40,000	1.22 9.06
Nitrogen dioxide	Annual	100	3.11
PM_{10}	Annual 24 hours (interim) 24 hours (99th percentile over 3 years)	50 150 150	Less than 0.01 0.11 Not available
$PM_{2.5}$	3-year annual 24 hours (98th percentile over 3 years)	15 65	Not available Not available
Sulfur dioxide	Annual 24 hours 3 hours	80 365 1,300	Less than 0.01 0.12 0.91
State-regulated Pollutants			
Gaseous fluoride	30 days 7 days 24 hours 12 hours	0.8 1.6 2.9 3.7	0.01 0.03 0.06 0.11
Total suspended particulates	Annual	75	Less than 0.01
Hazardous/Toxic Compounds			
1,1,1-trichloroethane	24 hours	9,550	Less than 0.01
Benzene	24 hours	150	0.01
Ethanolamine	24 hours	200	Less than 0.01
Ethyl benzene	24 hours	4,350	Less than 0.01
Ethylene glycol	24 hours	650	Less than 0.01
Formaldehyde	24 hours	15	Less than 0.01
Glycol ethers	24 hours	No standard	Less than 0.01
Hexachloronaphthalene	24 hours	1	Less than 0.01
Hexane	24 hours	900	0.01
Manganese	24 hours	25	Less than 0.01
Methyl alcohol	24 hours	1,310	Less than 0.01
Methyl-ethyl-ketone	24 hours	14,750	Less than 0.01
Methyl-isobutyl-ketone	24 hours	2,050	Less than 0.01
Methylene chloride	24 hours	8,750	0.01
Naphthalene	24 hours	1,250	Less than 0.01
Nitric acid	24 hours	125	0.28
Phenol	24 hours	190	Less than 0.01
Phosphorous	24 hours	0.5	Less than 0.01
Sodium hydroxide	24 hours	50	Less than 0.01
Toluene	24 hours	2,000	0.01
Trichloroethane	24 hours	6,750	Less than 0.01
Vinyl acetate	24 hours	176	Less than 0.01
Xylene	24 hours	4,350	0.02

PM_n = Particulate matter less than or equal to *n* microns in diameter.
Source: Bickford et al. 1997.

Radiological Gaseous Emissions

The decladding and cleaning of blanket spent nuclear fuel and the electrometallurgical treatment of driver spent nuclear fuel at ANL-W would release gaseous fission products to the hot (argon) cell environment. Krypton-85 and elemental tritium are the most prevalent radioactive gaseous fission products that would be released to the environment. The released tritium in the cell would not be oxidized due to a very low presence of oxygen and humidity in the argon cell. The argon cell also contains an equilibrium concentration of other radionuclide isotopes. Appendix E, Section E.4.1, provides a list of various isotopes that are present in the argon cell in nanocuries (10^{-9} curies) and are released to the atmosphere through the facility stack, along with the krypton-85 and elemental tritium. The maximum release of radioactive gaseous emissions would occur when decladding blanket spent nuclear fuel for packaging and shipment to SRS and electrometallurgical treatment of driver spent nuclear fuel are performed simultaneously. This simultaneous operation was estimated to occur over a three-year period starting in 2003. Appendix E, Section E.4.2, provides details on releases during the processing period at ANL-W. Based on an annual decladding throughput of 10 metric tons of heavy metal of blanket spent nuclear fuel elements and an electrometallurgical treatment processing of about 0.6 metric tons of heavy metal of driver spent nuclear fuel elements, about 809 curies of elemental tritium and 11,860 curies of gaseous krypton-85 would be released annually to the atmosphere (see Appendix E, Section E.4.2).

Since declad and clean fuel would be packaged and sent to SRS, some gaseous fission products would be expected to be present in that fuel. However, it was assumed conservatively that all gaseous fission products in the blanket spent nuclear fuel would be released to the environment during PUREX processing at SRS over a six-month period (see Appendix E, Section E.4.3). The radiological exposures to the public and workers from radioactive emissions are presented in Section 4.5.4.

4.5.2 Water Resources

As stated in Section 4.4.2, decladding and cleaning of blanket spent nuclear fuel and treatment of driver spent nuclear fuel using electrometallurgical treatment would not discharge any radiological chemical material to the surface or groundwater at the INEEL site. These activities also would not impact the current groundwater usage at the site. For a discussion of impacts on water resources at ANL-W, see Section 4.4.2.

The impacts on water resources from processing blanket spent nuclear fuel at F-Canyon are described below.

Surface Water

No surface water would be used for PUREX processing of blanket spent nuclear fuel at the F-Area. The F-Canyon processing facilities are outside the 100-year floodplain, as shown in Figure 3-6.

Nonradiological Liquid Effluent

The major sources of liquid effluent from PUREX processing of blanket spent nuclear fuel at SRS would be process cooling water and steam condensate. There are sufficient capacities in existing wastewater treatment facilities to handle the liquid effluent from this processing. Liquid effluent associated with PUREX processes would use these facilities and the existing permitted outfalls (Section 3.3.4.1). Process cooling water treatment would result in releases to Upper Three Runs Creek from the F-Area, as shown in **Table 4–26**. Sanitary waste would be treated at the SRS Central Wastewater Treatment Facility and discharged through an existing NPDES-permitted outfall. Since employment would not increase as a result of processing this fuel, the treatment rates through the Central Wastewater Treatment Facility would not be affected and the requirements of the SRS NPDES permit would continue to be met (DOE 2000).

Table 4–26 Chemical Effluent Concentrations From PUREX Cooling Water Treatment

Parameter	Effluent Concentrations	Existing Stream Water Concentrations		Water Quality Criterion (milligrams per liter) [c]
	F-Area (milligrams per liter)	Upper Three Runs (Upstream) [a] (Average) (milligrams per liter)	Upper Three Runs (Downstream) [b] (Average) (milligrams per liter)	
Aluminum	0.2	0.19	0.24	(d)
Ammonia	0.03	0.0001	Not reported	(d)
Chromium	0.02	Not detected	Not detected	0.1
Copper	0.01	0.018	0.015	1
Manganese	0.01	0.039	0.052	0.05
Nickel	0.05	Not detected	Not detected	0.1
Nitrate	0.04	0.36	0.27	10
Zinc	0.07	0.06	0.091	3

[a] Stream monitor U3R-1A.
[b] Stream monitor U3R-4.
[c] Federal Drinking Water Standards and Health Advisories (EPA 1996) and South Carolina Water Quality Criteria for Protection of Human Health (SCDHEC 1998).
[d] No drinking water standard.
Sources: Arnett and Mamatey 1998, DOE 2000.

Although proposed or final Federal drinking water standards do not apply to the discharges, these standards are used for comparison to SRS discharges. The discharge concentration would not exceed the Federal drinking water standard. The discharges would also comply with the South Carolina Water Classifications and Standards (SCDHEC 1998). The release concentrations would be no greater than the concentrations measured in Upper Three Runs (Arnett and Mamatey 1998), with the exception of zinc and ammonia. Zinc concentrations in the discharge are within the Federal health advisory limits (EPA 1996).

Radiological Liquid Effluent

PUREX processing would release measurable radioactive nuclides to the surface water through the cooling water system. The expected radiological effluent from processing declad and cleaned blanket spent nuclear fuel at F-Canyon was estimated based on the measured data from various effluent streams at F-Area as presented in the SRS Environmental Data for 1997 (Arnett and Mamatey 1998). Since the mechanism associated with releases of liquid effluent from PUREX processing at F-Canyon is essentially the same for almost every fuel type processed, the F-Area 1997 effluent data were used to conservatively represent the potential releases from a six-month operation of F-Canyon. Table 4–27 provides a list of potential radiological isotopes that could be released to the surface water during processing of approximately 57 metric tons of heavy metal of blanket spent nuclear fuel (see Appendix E, Section E.4.3, for details).

Groundwater

All process water would come from groundwater, as would sanitary water. At most, less than 65 million liters (17 million gallons) per year would be required for cooling water. SRS annually withdraws more than 5 billion liters (more than 1.3 billion gallons) per year of groundwater (DOE 2000).

Table 4–27 Estimated Radiological Liquid Effluent From PUREX Processing of Blanket Spent Nuclear Fuel

Isotope	*Curies*
Tritium (Hydrogen-3)	1.54
Strontium-89/Strontium-90	0.000031
Cesium-137	0.0022
Uranium-234	0.000085
Promethium-147	0.000011
Uranium-238	0.00019
Plutonium-238	0.000016
Plutonium-239	7.8×10^{-6}

Source: Arnett and Mamatey 1998.

Nonradiological Liquid Effluent

No nonradiological chemicals would be discharged to groundwater from PUREX processing of blanket spent nuclear fuel at F-Canyon and the FB-Line in F-Area.

Radiological Liquid Effluent

No radiological liquid effluent or waste would be discharged to groundwater from PUREX processing of blanket spent nuclear fuel at F-Canyon and the FB-Line in F-Area.

4.5.3 Socioeconomics

Under this alternative, the existing facilities at ANL-W and SRS would remain operational. No new employment or in-migration of workers would be required. Thus, there would be no additional impacts on the socioeconomic conditions in the regions around INEEL and SRS.

4.5.4 Public and Occupational Health and Safety

The assessments of potential incremental radiological and chemical impacts associated with this alternative are presented in this section. Summaries of radiological impacts from normal operations are presented in Tables 4–28 and 4–29 for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in Tables 4–31 through 4–34. The impacts from hazardous chemical releases during accident conditions are presented in Tables 4–35 and 4–36. Background information on the effects of radiation on human health and safety is presented in Appendix E, Section E.2.

4.5.4.1 Normal Operations

Radiological Impacts

Under this alternative, radioactive releases would occur during PUREX processing at F-Canyon. Appendix E, Sections E.3 and E.4.3, details the method and assumptions used for calculating the impacts of normal operational radiological releases on the public health and safety. Doses to the public would result from treating about 57 metric tons of heavy metal of blanket spent nuclear fuel. The blanket spent nuclear fuel being processed at SRS is already declad and cleaned at ANL-W; therefore, the gaseous fission products are assumed to have already been released. However, for the analytical purposes of this EIS, it was conservatively assumed

that the gaseous fission products would still be within the matrix of the fuel and would be released during PUREX processing at SRS. The processing was assumed to continue for six months (see Appendix E.4.3).

Calculated incremental maximum annual and project total radiological impacts to the public are given in **Table 4–28**. The impacts are calculated for two types of receptors: the general public living within 80 kilometers (50 miles) of F-Canyon in the year 2010, and a maximally exposed offsite individual (a member of the public assumed to be residing at the SRS site boundary and receiving the maximum dose). Since PUREX processing would produce radiological air emissions as well as radiological liquid effluent, doses to the public were calculated considering both the air emissions and liquid effluent. Primary contributors to public doses would be from tritium gas (assumed to be tritium oxide) and krypton-85, which together contribute over 95 percent of the total calculated doses. The doses resulting from liquid effluent were estimated from data provided in support of the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000) (see Appendix E, Section E.4.3, for details). The doses and duration from decladding and cleaning blanket spent nuclear fuel and treatment of driver spent nuclear fuel at ANL-W would be similar to those presented for Alternative 2 in Section 4.4.4.1. To put the impacts into perspective, comparisons to natural background radiation levels are included in Table 4–28.

Table 4–28 Annual and Project Total Radiological Impacts to the Public From Operational Activities Under Alternative 3

Receptor	*PUREX Process Declad and Cleaned Blanket Spent Nuclear Fuel at SRS* [a,b,c]	*Declad and Clean Blanket Spent Nuclear Fuel and Electrometallurgically Treat Driver Spent Nuclear Fuel at ANL-W*
Population Within 80 Kilometers (50 Miles) in the Year 2010		
Collective dose (person-rem per year) [d]	0.020	0.0030
Excess latent cancer fatalities (per year)	0.000010	1.5×10^{-6}
Project total excess latent cancer fatalities [f]	0.000010	8.3×10^{-6}
Maximally Exposed Offsite Individual		
Dose (millirem per year) [d]	0.00051	0.00038
Percent of annual background radiation [e]	0.00017	0.00011
Latent cancer fatality risk (per year)	2.6×10^{-10}	1.9×10^{-10}
Project total lifetime cancer fatality risk [f]	2.6×10^{-10}	1.0×10^{-9}
Average Individual Within 80 Kilometers (50 Miles)		
Dose (millirem per year) [g]	0.000024	0.000012
Latent cancer fatality risk (per year)	1.2×10^{-11}	6.2×10^{-12}
Project total lifetime cancer fatality risk [f]	1.2×10^{-11}	3.5×10^{-11}

[a] Includes dose from air emissions and liquid effluent over the six-month processing duration.
[b] Doses to the population and the maximally exposed offsite individual from liquid effluent are 0.00068 person-rem and 0.00012 millirem, respectively.
[c] Since PUREX operations would last less than one year, the values of the project total dose and risk are equal to the corresponding annual values.
[d] Annual maximum dose during normal operations.
[e] The annual natural background radiation level at INEEL and at SRS is about 360 and 300 millirem, respectively, for the average individual (see Tables 3–8 and 3–20); the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 person-rem at INEEL and 254,000 person-rem at SRS.
[f] Total calculated risk over nine years at ANL-W and six months at SRS.
[g] Obtained by dividing the population dose by the number of people projected to live in the year 2010 within 80 kilometers (50 miles) of ANL-W (240,338) and SRS F-Canyon (848,000).

As shown in this table, the expected radiation doses to the maximally exposed offsite individual and the general public would be much smaller than the limit of 10 millirem per year set by the EPA (40 CFR 61) and DOE (DOE Order 5400.5).

Table 4–29 summarizes worker population doses. Occupational doses were estimated by examining the type and duration of various operations performed by SRS workers involved with the PUREX process. The estimated annual total worker population dose would be 75 person-rem, with an average individual dose of 500 millirem per year for each of the 150 involved workers. If these estimates were projected for six months of PUREX activities, the project total worker population dose would be 38 person-rem, leading to a risk of 0.015 latent cancer fatalities. The estimated annual total worker population dose to treat driver spent nuclear fuel at ANL-W is 22 person-rem, as indicated in Section 4.4.4.1.

Table 4–29 Annual and Project Total Radiological Impacts to Workers From Operational Activities Under Alternative 3

Receptor	*Impacts*	
Worker [a]	Operations at SRS	Operations at ANL-W
Average worker dose (millirem per year)	250 [b]	60
Average worker latent cancer fatality risk (project total)	0.00010 [b]	0.00022 [c]
Worker Population		
Collective dose (person-rem per year)	38 [b]	22
Excess latent cancer fatalities (per year)	0.015 [b]	0.0088
Project total dose (person-rem)	38 [b]	231 [c]
Project total excess latent cancer fatalities	0.015 [b]	0.092 [c]

[a] The regulatory dose limit for an individual worker is 5,000 millirem per year (10 CFR 835). However, the maximum annual dose to a worker would be kept below the DOE Administrative Control Level of 2,000 millirem per year, as established for all DOE activities in DOE Order N 441.1.

[b] Operations at SRS to treat blanket spent nuclear fuel at F-Canyon are performed over six months.

[c] Operations at ANL-W to declad and clean blanket spent nuclear fuel and treat driver spent nuclear fuel are performed over nine years plus one year for deactivation of processing facilities; see Section 4.4.1.

Sources: ANL 1999, DOE 2000.

As shown in Tables 4–28 and 4–29:

- The annual dose to the maximally exposed offsite individual at ANL-W would be 0.00038 millirem per year, with an associated risk of developing a fatal cancer of 1.9×10^{-10} per year (or one chance in 5.3 billion that the individual would develop a fatal cancer per year of exposure).

- The collective dose to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 0.0030 person-rem per year, with an associated 1.5×10^{-6} latent cancer fatalities per year (or one chance in 670,000 that the population would experience a fatal cancer per year of exposure).

- The project total risk to the population within 80 kilometers (50 miles) of the ANL-W site would be 8.3×10^{-6} latent cancer fatalities (or one chance in 120,000 that the exposed population would experience a fatal cancer).

- The collective dose to ANL-W facility workers would be 22 person-rem per year, with an associated 0.0088 latent cancer fatalities per year (or one chance in 113 that the workers would experience a fatal cancer per year of operation).

- The project total dose to ANL-W facilities workers would be 231 person-rem, with an associated 0.092 latent cancer fatalities (or one chance in 11 that the exposed workers would experience a fatal cancer).

- The project total dose to the maximally exposed offsite individual at SRS from six-month PUREX processing would be 0.00051 millirem, with an associated risk of developing a fatal cancer of 2.6×10^{-10} (or one chance in 3.8 billion that the individual would develop a fatal cancer).

- The project total dose to the population within 80 kilometers (50 miles) of the F-Canyon would be 0.020 person-rem, with an associated 0.000010 latent cancer fatalities (or one chance in 100,000 that the exposed population would experience a fatal cancer).

- The project total dose to F-Canyon facility workers would be 38 person-rem, with an associated 0.015 latent cancer fatalities (or one chance in 67 that the exposed workers would experience a fatal cancer).

These results indicate that no increase in the expected number of fatal cancers among the various affected populations would result from the activities under this alternative.

Hazardous Chemical Impacts

The hazardous chemical impacts to the public residing at the INEEL site boundary from the operational activities at ANL-W under Alternative 3 would be similar to the impacts evaluated for Alternative 1 described in Section 4.3.4.1. The results indicate that no adverse toxic health or cancer effects would be expected from exposure to hazardous chemicals released under this alternative. The existing baseline chemical environment is presented in Section 3.2.10.2.

For SRS, both carcinogenic and noncarcinogenic health effects to the public were assessed from exposure to hazardous chemicals; the results are summarized in **Table 4–30.**

Table 4–30 Hazardous Chemical Impacts to the Public From Operational Activities at SRS Under Alternative 3

Chemical	*Annual Concentration (milligrams per cubic meter)*	*Hazard Quotient (noncarcinogenic chemicals)*	*Cancer Risk (carcinogenic chemicals)*
Benzene	1.4×10^{-6}	None	1.1×10^{-8}
Ethyl benzene	1.3×10^{-6}	1.3×10^{-6}	None
Formaldehyde	1.3×10^{-6}	None	1.6×10^{-8}
Hexane	1.4×10^{-6}	7.1×10^{-6}	None
Manganese	1.3×10^{-6}	0.025	None
Methyl ethyl ketone	2.5×10^{-6}	2.5×10^{-6}	None
Methylene chloride	7.1×10^{-7}	None	3.3×10^{-10}
Naphthalene	1.3×10^{-6}	0.00042	None
Toluene	1.4×10^{-6}	3.5×10^{-6}	None
Vinyl acetate	1.3×10^{-6}	6.3×10^{-6}	None
Hazard Index		0.025	Not applicable

The results indicate that no adverse toxic (noncarcinogenic) health effects or cancer potency are expected from exposure to hazardous chemicals released at SRS under this alternative. See Appendix E, Section E.5, for more details. The existing baseline chemical environment is presented in Section 3.3.10.2.

4.5.4.2 Facility Accidents

Radiological Impacts

Potential radiological impacts to the public and a noninvolved onsite worker due to accidents during operational activities associated with decladding, cleaning, and PUREX processing of blanket spent nuclear fuel and electrometallurgical treatment of driver spent nuclear fuel are summarized and presented in this section. The detailed analysis of facility accidents, with the associated assumptions, is presented in Appendix F. The detailed analysis considered a wide spectrum of potential accident scenarios, including fire, spills, criticality, earthquake, and aircraft crash. The aircraft crash and criticality events were determined to have an occurrence frequency of less than 10^{-7} per year, and consequence analyses for these two events were not performed. Decladding and cleaning of blanket spent nuclear fuel would be performed in the Hot Fuel Examination Facility; treatment of driver spent nuclear fuel would be performed in both the Hot Fuel Examination Facility and the Fuel Conditioning Facility. Because driver spent nuclear fuel processing would take place in both of these facilities, the beyond-design-basis earthquake event is assessed for the driver spent nuclear fuel, taking into account the multifacility impacts of this event. Decladding and cleaning blanket spent nuclear fuel would be performed only in the Hot Fuel Examination Facility. The multifacility impacts of the beyond-design-basis earthquake are not relevant to the blanket spent nuclear fuel. Therefore, only the higher frequency design-basis earthquake event was analyzed for the blanket spent nuclear fuel. **Table 4–31** presents the frequencies and consequences of the postulated set of accidents at ANL-W to the maximally exposed offsite individual, the offsite population residing within 80 kilometers (50 miles) of the facility, and a noninvolved worker located 100 meters (330 feet) to 230 meters (755 feet) from the facility.

Table 4–31 Accident Frequency and Consequences at ANL-W Under Alternative 3

Accident	Frequency (event per year)	Maximally Exposed Offsite Individual		Population Within 80 Kilometers (50 Miles)		Noninvolved Worker	
		Dose (millirem)	Latent Cancer Fatality Risk [a]	Dose (person-rem)	Excess Latent Cancer Fatalities [b]	Dose (millirem)	Latent Cancer Fatality Risk [a]
Driver Spent Nuclear Fuel							
Salt powder spill	0.01	0.00046	2.3×10^{-10}	0.000098	4.9×10^{-8}	4.7×10^{-7}	1.9×10^{-13}
Salt transfer drop	1.0×10^{-7}	0.19	9.5×10^{-8}	0.022	0.000011	0.073	2.9×10^{-8}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Cask drop	0.01	0.030	1.5×10^{-8}	0.0035	1.8×10^{-6}	0.00084	3.4×10^{-10}
Design-basis earthquake	0.008	12	6.0×10^{-6}	1.4	0.0007	4.7	1.9×10^{-6}
Beyond-design-basis earthquake	0.00001	22,000	0.022	2,500	1.3	370	0.00015
Blanket Spent Nuclear Fuel							
Cask drop	0.01	0.0024	1.2×10^{-9}	0.00028	1.4×10^{-7}	0.000049	2.0×10^{-11}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Sodium fire [c]	0.008	5.9	3.0×10^{-6}	0.69	0.00035	0.054	2.2×10^{-8}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.
[c] The frequency for this accident is the frequency for the facility design-basis earthquake initiating cell fire.

The dose to the maximally exposed offsite individual was calculated for the 95th percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50th percentile meteorological conditions. DOE did not quantitatively estimate the involved worker dose due to accidents (see the discussion on the involved worker in Section 4.2.4.2). The accident risks for the same receptors are summarized in **Table 4–32**.

Table 4–32 Annual Cancer Risks Due to Accidents at ANL-W Under Alternative 3

Accident	*Maximally Exposed Offsite Individual* [a]	*Population Within 80 Kilometers (50 Miles)* [b]	*Noninvolved Worker* [a]
Driver Spent Nuclear Fuel			
Salt powder spill	2.3×10^{-12}	4.9×10^{-10}	1.9×10^{-15}
Salt transfer drop	9.5×10^{-15}	1.1×10^{-12}	2.9×10^{-15}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Cask drop	1.5×10^{-10}	1.7×10^{-8}	3.4×10^{-12}
Design-basis earthquake	4.8×10^{-8}	5.6×10^{-6}	1.5×10^{-8}
Beyond-design-basis earthquake	2.2×10^{-7}	0.000013	1.5×10^{-9}
Blanket Spent Nuclear Fuel			
Cask drop	1.2×10^{-11}	1.4×10^{-9}	2.0×10^{-13}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Sodium fire	2.4×10^{-8}	2.8×10^{-6}	1.7×10^{-10}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

For accidents at ANL-W, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be 2.2×10^{-7} per year (or one chance in 4.5 million that the individual would develop a fatal cancer per year of operation) and 1.5×10^{-8} per year (or one chance in 66.7 million that the individual would develop a fatal cancer per year of operation), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.000013 per year (or one chance in 76,920 that the population would experience a fatal cancer per year of operation).

The potential radiological impacts to the public and a noninvolved onsite worker due to accidents during PUREX operational activities at SRS are summarized below. The detailed analysis of facility accidents, with the associated assumptions, is presented in Appendix F. **Table 4–33** presents the frequencies and consequences of the postulated set of accidents to the maximally exposed offsite individual, the offsite population residing within 80 kilometers (50 miles) of the facility, and a noninvolved worker located 100 meters (330 feet) to 350 meters (1150 feet) from the facility. The 350-meter (1150-foot) distance leads to a higher dose to the noninvolved worker for the scenarios with elevated releases.

The dose to the maximally exposed offsite individual was calculated for the 95th percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50th percentile meteorological conditions. DOE did not quantitatively estimate the facility worker population dose due to accidents. The accident risks for the same receptors are summarized in **Table 4–34**.

Table 4–33 Accident Frequency and Consequences at SRS Under Alternative 3

Accident	Frequency (event per year)	Maximally Exposed Offsite Individual		Population Within 80 Kilometers (50 Miles)		Noninvolved Worker	
		Dose (millirem)	Latent Cancer Fatality Risk [a]	Dose (person-rem)	Excess Latent Cancer Fatalities [b]	Dose (millirem)	Latent Cancer Fatality Risk [a]
Fire (F-Canyon)	0.000061	610	0.00031	5500	2.8	2300	0.00092
Explosion (FB-Line)	0.00010	6.5	3.3×10^{-6}	53	0.027	19	7.6×10^{-6}
Design-basis earthquake (F-Canyon)	0.00013	1100	0.00055	2100	1.1	12000	0.0048
Design-basis earthquake (FB-Line)	0.00013	58	0.000029	120	0.06	900	0.00036
Criticality	0.00010	11	5.5×10^{-6}	59	0.030	37	0.000015

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

Table 4–34 Annual Cancer Risks Due to Accidents at SRS Under Alternative 3

Accident	Maximally Exposed Offsite Individual [a]	Population Within 80 Kilometers (50 Miles) [b]	Noninvolved Worker [a]
Fire (F-Canyon)	1.9×10^{-8}	0.00017	5.6×10^{-8}
Explosion (FB-Line)	3.3×10^{-10}	2.7×10^{-6}	7.6×10^{-10}
Design-basis earthquake (F-Canyon)	7.2×10^{-8}	0.00014	6.2×10^{-7}
Design-basis explosion (FB-Line)	3.8×10^{-9}	7.8×10^{-6}	4.7×10^{-8}
Criticality	5.5×10^{-10}	3.0×10^{-6}	1.5×10^{-9}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

For accidents at SRS, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be 7.2×10^{-8} per year (or one chance in 13.9 million that the individual would develop a fatal cancer per year of operation) and 6.2×10^{-7} per year (or one chance in 1.6 million that the individual would develop a fatal cancer per year of operation), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.00017 per year (or one chance in 5,880 that the population would experience a fatal cancer per year of operation).

Hazardous Chemical Impacts

Nonradiological impacts are evaluated in terms of comparison to ERPG. ERPG values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects (see Appendix F, Section F.3.1.2, for details).

The hazardous chemical impacts of potential facility accidents at ANL-W associated with the treatment of driver spent nuclear fuel using electrometallurgical treatment are summarized in **Table 4–35**.

Table 4–35 Hazardous Chemical Impacts Due to Accidents at ANL-W Under Alternative 3

Accident	*Frequency (event per year)*	*Receptor*	*Exposure*
Uranium handling accident	0.01	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Design-basis earthquake	0.0002	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Sodium fire	0.008	Noninvolved worker	Sodium: less than ERPG-1
		Maximally exposed offsite individual	Sodium: less than ERPG-1
Uranium fire	0.00001	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Beyond-design-basis earthquake	0.00001	Noninvolved worker	Cadmium: less than ERPG-1 Uranium: less than ERPG-1
		Maximally exposed offsite individual	Cadmium: less than ERPG-1 Uranium: less than ERPG-1

ERPG = Emergency Response Planning Guideline.

The SRS Spent Nuclear Fuel Management Final EIS (DOE 2000) analyzed the consequences of three chemical spills involving hazardous chemicals in the F-Area: (1) the loss of 50 percent sodium hydroxide containment from a skid-mounted 1,000-gallon dumpster; (2) the loss of 50 percent nitric acid containment from a skid-mounted 1,000-gallon dumpster; and (3) the loss of 30 percent sodium nitrite containment from a skid-mounted 1,000-gallon dumpster and an adjacent 1,600-gallon holdup tank. These analyses are summarized in the **Table 4–36**, and are considered representative of wet storage accidents at SRS.

Table 4–36 Hazardous Chemical Impacts Due to Accidents at SRS Under Alternative 3

Accident	*Frequency (Event Per Year)*	*Receptor*	*Exposure*
Wet storage, container rupture	0.005	Noninvolved worker	Sodium hydroxide: less than Permissible Exposure Limit-Time Weighted Average
Wet storage, container rupture	0.005	Noninvolved worker	Nitric acid: less than Permissible Exposure Limit-Time Weighted Average
		Maximally exposed offsite individual	Nitric acid: less than Permissible Exposure Limit-Time Weighted Average
Wet storage, container rupture	0.006	Noninvolved worker	Sodium nitrite: less than Permissible Exposure Limit-Time Weighted Average

Permissible Exposure Limit-Time Weighted Average is used for chemicals having no ERPG values. It is considered to be less than ERPG-1.

Source: DOE 2000.

4.5.5 Environmental Justice

As discussed in Appendix H, Executive Order 12898 directs Federal agencies to address disproportionately high and adverse health or environmental effects of alternatives on minority or low-income populations.

Analyses of normal operations and accident conditions presented in Section 4.3.4 show the risk of latent cancer fatalities to the public residing within 80 kilometers (50 miles) of the electrometallurgical treatment and decladding and cleaning processing facilities at ANL-W and the PUREX processing facility at SRS to be much

lower than 1. Therefore, radiological and nonradiological risks posed by implementation of this alternative would not result in disproportionately high and adverse consequences to any particular group within the general population, including minority or low-income populations.

4.5.6 Waste Management

ANL-W

This alternative would generate process waste from treatment operations, other associated process waste from normal support operations, and deactivation waste following the conclusion of operations. Process waste would include fuel assembly hardware and high-level radioactive metallic and ceramic waste. Other associated process waste would include operational waste such as failed equipment, rags, packaging materials, and other miscellaneous items. Deactivation waste would include the disposal of process equipment and other materials. All of these materials would be categorized according to existing DOE Orders and ANL-W waste management procedures. The anticipated categorization of these waste types generated at ANL-W and their expected interim storage and final disposal locations are given in Table 4–9 (see Section 4.2.6). The quantities of various waste forms generated as a result of Alternative 3 at ANL-W are provided in **Table 4–37.**

Estimates of the total amount of other associated process waste that would be generated are based on an evaluation of waste forecasts from ANL-W that accounted only for the fraction of total ANL-W waste that would be attributable to the treatment of sodium-bonded spent nuclear fuel. The values in Table 4–37 are for disposal and account for volume reduction. It is anticipated that a large fraction of the low-level radioactive waste generated as a result of Alternative 3 could be volume-reduced at the Waste Experimental Reduction Facility at INEEL, prior to disposal at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex.

Table 4–37 Amounts of Waste Generated at ANL-W Under Alternative 3[a]

Waste Stream	*Waste Quantities*	
	Volume (cubic meters)	*Mass (kilograms)*
Direct Process Waste		
Fuel assembly hardware (low-level radioactive waste)	37.5	13,100
High-level radioactive ceramic waste	16.3 (26 canisters)[b]	24,400
High-level radioactive metallic waste	1.3 (2 canisters)[b]	2,500
Spent nuclear fuel	0	0
Other Associated Process Waste		
High-level radioactive waste	0.4 (1 canister)[b]	220
Low-level radioactive waste[c]	555	113,000
Transuranic waste	9.1	3,800
Mixed waste	27.5	14,800
Sanitary waste	4,960	1.72×10^6
Deactivation Waste		
Low-level radioactive waste[c]	178	60,000
Transuranic waste	1.6	853
Mixed waste	5.1	3,400

[a] These waste generation estimates are through the year 2015. This is the assumed date that these materials might be sent to the repository. Treatment, high-level radioactive waste processing, deactivation, and interim storage would be accomplished during this time period.

[b] Standardized canisters.

[c] The volumes listed represent final disposal volumes following volume reduction at the Waste Experimental Reduction Facility at INEEL.

Source: ANL 1999.

The waste values in Table 4–37 are total quantities that would be produced as a result of Alternative 3 operations at ANL-W. They are not incremental increases over the volumes provided in Table 4–10 that would result from the No Action Alternative. In Alternative 3, the driver spent nuclear fuel would be transformed into high-level radioactive waste forms (ceramic and metallic) at ANL-W for disposal in the repository, and in this conversion process, the total volume of material to be disposed of in the repository would be reduced from direct disposal values. The blanket spent nuclear fuel would be cleaned and declad and sent to SRS for PUREX processing. The high-level radioactive waste that would be generated from PUREX processing at SRS is presented in Table 4–39.

Direct Process Waste

For this alternative, fuel hardware would be removed from the fuel elements in the Fuel Conditioning Facility air cell and disposed of as low-level radioactive waste. These components are primarily stainless steel materials that contain short-lived radionuclides. This waste stream has been produced at ANL-W for many years and would be handled, as in the past, according to DOE Orders and ANL-W waste management procedures. In addition, the blanket spent nuclear fuel cladding is included in the fuel hardware stream.

Under this alternative, metallic and ceramic high-level radioactive waste would be a primary product of the electrometallurgical treatment of driver spent nuclear fuel. The salt removed from the electrorefiners would contain the majority of fission products and transuranics from the spent nuclear fuel. This removed salt would be packaged and transferred to the Hot Fuel Examination Facility for processing into ceramic waste. The metallic waste form would consist primarily of stainless steel cladding hulls containing the noble metal fission products. The hulls would be removed from the electrorefiner and packaged for shipment to the Hot Fuel Examination Facility for processing into the metallic waste form. Both the ceramic and metallic waste would be categorized as high-level radioactive waste. The volumes of waste forms provided in Table 4–37 are for the standardized canisters required for disposal of these materials.

The metallic and ceramic high-level radioactive waste that would be generated as a result of electrometallurgical treatment of driver spent nuclear fuel under this alternative and the driver and blanket spent nuclear fuel generated during the demonstration project at ANL-W would be stored temporarily at the Radioactive Scrap and Waste Facility at ANL-W to allow retrieval for future disposal. The Radioactive Scrap and Waste Facility was designed and constructed for temporary storage of this type of waste. Shielding would be provided by a combination of (1) steel storage liners which would store the waste, and (2) the soil surrounding the liners. When a geologic repository is available, the waste cans containing the metallic and ceramic high-level radioactive waste would be removed from storage, shipped to the INEEL Dry Transfer Facility, and prepared for shipment to the repository.

Other Associated Process High-Level Radioactive Waste

High-level radioactive waste could be generated as a result of blanket spent nuclear fuel processing at ANL-W. This would result from activities in the Hot Fuel Examination Facility. Material in this waste stream would consist of the absorbent used in the off-gas system which has collected the volatile radionuclides released from the spent nuclear fuel when heated.

The volume of high-level radioactive waste that would be generated is expected to be less than the amount needed to fill a single high-level radioactive waste canister. Conservatively the volume of a single canister, 0.4 cubic meters, has been used for the volume of high-level radioactive waste generated.

Other Associated Process Low-Level Radioactive Waste

Low-level radioactive waste would be generated as a result of decladding and cleaning blanket spent nuclear fuel and treatment of driver spent nuclear fuel at ANL-W. This would result from activities in the Fuel Conditioning Facility and the Hot Fuel Examination Facility at ANL-W (e.g., equipment decontamination and repair), as well as in other facilities at ANL-W (e.g., analytical laboratory activities). Material in this waste stream has been generated and routinely handled at ANL-W for many years.

The volume of low-level radioactive waste resulting from decladding and cleaning blanket spent nuclear fuel and electrometallurgical treatment of driver spent nuclear fuel at ANL-W that would require disposal (after volume reduction) would be approximately 50 cubic meters (1,766 cubic feet) per year. This represents approximately 1 percent of the total annual volume of low-level radioactive waste currently being disposed of at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex, and the total of 555 cubic meters (19,600 cubic feet) represents approximately 0.5 percent of the total Radioactive Waste Management Complex disposal inventory.

Other Associated Process Transuranic Waste

Transuranic waste would be generated by Alternative 3 from decontamination activities for repair and maintenance of items, and miscellaneous work associated with processing the sodium-bonded spent nuclear fuel. Transuranic waste would be generated primarily from activities conducted in gloveboxes and hot cells at ANL-W.

All of the transuranic waste generated would be packaged and certified in accordance with Waste Isolation Pilot Plant Acceptance Criteria prior to transport to the Waste Isolation Pilot Plant facility. The transuranic waste that would be generated would amount to approximately 1 cubic meter (35 cubic feet) per year, which is less than 0.002 percent of the volume of transuranic waste in retrievable storage at the Radioactive Waste Management Complex at INEEL. The total volume of transuranic waste is 9.1 cubic meters (321 cubic feet), which is less than 0.005 percent of the estimated total volume of transuranic waste to be placed at the Waste Isolation Pilot Plant.

Other Associated Process Mixed Waste

Mixed waste would be generated primarily from the disposal of any cadmium-contaminated equipment or cleanup material and the analysis of cadmium samples. Mixed waste would be handled according to ANL-W procedures that require limited accumulation at the point of generation. Interim storage of this waste would be accomplished at the Radioactive Scrap and Waste Facility prior to eventual disposal. The Radioactive Scrap and Waste Facility is a permitted mixed waste storage facility for these materials. The mixed waste streams that contribute to the overall mixed waste generated by electrometallurgical treatment have been identified in the INEEL Site Treatment Plan (DOE 1995b).

Deactivation Waste

A variety of waste would be generated as part of deactivation activities associated with decladding and cleaning blanket spent nuclear fuel and the treatment of driver spent nuclear fuel at ANL-W. This would include process equipment and process material, such as electrorefiner cadmium from electrometallurgical treatment of driver spent nuclear fuel. Generated waste categories would include low-level radioactive waste, transuranic waste, and mixed waste. This waste would be categorized and disposed of according to DOE Orders and ANL-W radioactive waste management procedures, as described above for each waste category.

The largest volume of deactivation waste would be low-level radioactive waste generated as a result of equipment dismantling and disposal. Components of electrometallurgical treatment of the driver spent nuclear fuel that would require disposition include two electrorefiners, two hot isostatic presses, and two V-mixers, as well as other components such as the grinder/crusher. Decontamination of these components would generate additional mixed, transuranic, and low-level radioactive waste that would require management. The deactivation waste volume is generated over two years. The total deactivation waste would represent an additional 30 percent over the total associated process waste (excluding sanitary waste) requiring disposal.

SRS

The PUREX process at SRS would generate process waste from treatment operations and other associated process waste from support operations. Process waste would include high-level radioactive waste. Other associated process waste would include operational waste such as failed equipment, rags, packaging materials, and other miscellaneous items. The associated process waste includes low-level radioactive waste, transuranic waste, and mixed waste. All of the waste streams would be categorized according to existing DOE Orders and SRS waste management procedures. The anticipated categorization of the waste types and their expected interim storage and final disposal locations are given in **Table 4–38**.

Table 4–38 Waste Material Categories at SRS and Interim and Final Locations

Waste Stream	*Category*	*Interim Storage Location*	*Final Disposal Location*
Process Waste			
Liquid waste form	High-level radioactive waste	Initial storage in the high-level radioactive waste Tank Farm followed by post-process storage at the Defense Waste Processing Facility.	Geologic repository
Other Associated Process Waste			
Less than 100 nanocuries per gram transuranic waste	Low-level radioactive waste	None	Low-activity waste vaults
Greater than 100 nanocuries per gram transuranic waste	Transuranic waste	Transuranic waste storage pads	Waste Isolation Pilot Plant
Contaminated	Mixed waste	Mixed waste storage buildings	Offsite

Estimates of the amounts of waste that would be generated as a result of the PUREX processing at SRS are provided in **Table 4–39**. These values are based on an evaluation of waste forecasts that account only for the fraction of total waste that would be attributable to processing the blanket spent nuclear fuel pins.

As indicated in the following waste type discussions, the amounts of waste associated with this alternative are relatively small compared to onsite and offsite management capacities.

Direct Process Waste

During the PUREX process, liquid high-level radioactive waste would be produced (along with plutonium metal and uranium solution). The liquid waste would be processed in the Defense Waste Processing Facility to yield vitrified high-level radioactive waste (borosilicate glass) and saltstone. This high-level radioactive waste would be temporarily stored at the Defense Waste Processing Facility pending ultimate disposal in a geologic repository. The saltstone is a cement form low-level radioactive waste that is generated or a by-product of SRS tank farm operations. The saltstone would be disposed of on site in the Z-Area Saltstone Vaults. The volume of this saltstone would be about 0.12 percent of the 1.11 million cubic meters (39.2 million cubic feet) storage capacity of the vaults.

Table 4–39 Amounts of Waste Generated at SRS Under Alternative 3

Waste Stream	*Waste Quantities (cubic meters)* [a]
Direct Process Waste	
Liquid high-level radioactive waste	510
Solid high-level radioactive waste [b,c]	5.6 (9 canisters) [b]
Saltstone [c]	1,290
Other Associated Process Waste	
Low-level radioactive waste	900 [d]
Transuranic waste	90
Mixed waste	6.9

[a] These values are estimated based on the heavy metal mass ratio of similar materials processed at SRS (20 metric tons of heavy metal) and provided in DOE 2000.
[b] Standardized high-level radioactive waste (Defense Waste Processing Facility) canisters.
[c] These waste forms result from processing the liquid high-level radioactive waste.
[d] Final disposal volume following a volume reduction (a reduction factor of 4 was assumed).

Other Associated Process Low-Level Radioactive Waste

Low-level radioactive waste would be generated during the PUREX process. The volume of low-level radioactive waste resulting from this alternative (after volume reduction) would be about 3 percent of the total 30,500-cubic meter (1.08 million-cubic foot) disposal capacity of the low-activity waste vaults.

Other Associated Process Transuranic Waste

The volume of transuranic waste that would be generated during the PUREX process would be only about 0.05 percent of the current 168,500-cubic meter (5.95 million-cubic foot) limit for the Waste Isolation Pilot Plant (DOE 1997).

Other Associated Process Mixed Waste

Mixed waste that would be generated during the PUREX process would be temporarily stored on site in the Mixed Waste Storage Buildings prior to eventual offsite disposal. The volume of this waste would be about 0.36 percent of the 1,900-cubic meter (67,100-cubic foot) storage capacity of these storage buildings.

4.6 Alternative 4: Melt and Dilute Blanket Fuel and Electrometallurgically Treat Driver Fuel at ANL-W

Under this alternative, the sodium-bonded blanket spent nuclear fuel would be cleaned to remove metallic sodium and then treated using the melt and dilute process at ANL-W. The melt and dilute product from treatment of this fuel would be stored at the Radioactive Scrap and Waste Facility pending repackaging and transportation for disposal in a geologic repository. The removed sodium would be stabilized using an oxidation/carbonation process (ANL 1999). The sodium-bonded driver spent nuclear fuel would be treated at ANL-W using the electrometallurgical treatment process. The process steps for the electrometallurgical treatment of driver spent nuclear fuel would be similar to those described earlier in Section 4.3 and in Appendix C. The treatment of driver spent nuclear fuel using the electrometallurgical process could start as early as 2000 and could be completed by 2006 to 2007. The preparation of blanket spent nuclear fuel could start in 2003, and subsequent melt and dilute treatment at ANL-W could start in 2005 and could be completed by 2012. In addition, a full year of operations would be needed to deactivate the processing equipment and establish an industrially safe configuration.

4.6.1 Air Quality

Nonradiological Gaseous Emissions

It is expected that criteria and hazardous air pollutants released from operational activities at ANL-W under this alternative would be the same as for Alternative 1, as described in Section 4.3.1 (see also Appendix E, Section E.5.3.1, for more detail). Baseline air quality concentrations are presented in Section 3.2.3.1.

Radiological Gaseous Emissions

Cleaning blanket spent nuclear fuel and electrometallurgically treating driver spent nuclear fuel would release gaseous fission products to the argon cell environment. Krypton-85 and elemental tritium are the most prevalent radioactive gaseous fission products that would be released to the environment. The tritium released into the cell would not be oxidized due to a very low presence of oxygen and humidity in the argon cell. The argon cell also contains an equilibrium concentration of other radionuclide isotopes. Appendix E, Section E.4.1, provides a list of various isotopes that are present in the argon cell in nanocuries (10^{-9} curies) and are released to the atmosphere through the facility stack, along with krypton-85 and elemental tritium. The maximum annual release of radioactive gaseous emissions would occur when electrometallurgical treatment processing of driver spent nuclear fuel is performed simultaneously with cutting blanket spent nuclear fuel for sodium removal prior to the melt and dilute process. This simultaneous operation would occur over a three-year period during the estimated 10 years of operation starting in 2003. Appendix E, Section E.4.2, provides details on releases during the processing period at ANL-W. Based on an annual blanket spent nuclear fuel processing (e.g., chopping and cleaning) throughput of 10 metric tons of heavy metal and electrometallurgical treatment processing of about 0.6 metric tons of heavy metal of driver spent nuclear fuel elements, about 809 curies of elemental tritium and 11,860 curies of gaseous krypton-85 could be released annually to the atmosphere (see Appendix E, Section E.4.2). The radiological exposures to the public and workers from radioactive emissions are presented in Section 4.6.4.

4.6.2 Water Resources

Surface Water

No surface water is used at ANL-W. Flood waters from the Big Lost River would not be expected to reach the facilities at ANL-W, as shown in Figure 3–3.

Nonradiological Liquid Effluent

There are no discharges to the surface waters at ANL-W, except for discharges of nonhazardous liquid waste to the sewage pond and to the industrial waste pond. Big Lost River, Little Lost River, and Birch Creek would not be impacted by activities associated with electrometallurgical and melt and dilute treatment processes. Current operating and monitoring practices would continue for stormwater and liquid effluent discharges associated with facilities at ANL-W (see Section 4.2.2).

During fuel treatment and associated activities, some hazardous materials may be used inside buildings. To prevent potential releases to surface or subsurface waters resulting from spills of hazardous materials used in buildings, these facilities are designed, constructed, and maintained to contain these materials. Double-contained pipes, leak detection, and secondary containment of tanks are some of the features used to prevent hazardous material releases to the environment. Following existing written procedures, spill containment and cleanup equipment is present in areas where hazardous materials are stored or used (DOE 1996b).

Radiological Liquid Effluent

No radiological liquid effluent generated by electrometallurgical and melt and dilute treatment process operations would be discharged to surface water.

Groundwater

Under this alternative at ANL-W, there would be little change in groundwater consumption for domestic use since there is little change expected in the number of workers. Water consumption for the electrometallurgical and melt and dilute treatment process operations would not impact the current water usage at ANL-W. The current water usage at ANL-W is 188 million liters (49.6 million gallons) per year.

Nonradiological Liquid Effluent

No nonradiological liquid effluent generated by electrometallurgical and melt and dilute treatment process operations would be discharged to groundwater.

Radiological Liquid Effluent

No radiological liquid effluent generated by electrometallurgical and melt and dilute treatment process operations would be discharged to groundwater.

4.6.3 Socioeconomics

Under this alternative, the existing facilities at ANL-W would remain operational. No new employment or in-migration of workers would be required. Thus, there would be no additional impacts on the socioeconomic conditions in the region around ANL-W and INEEL.

4.6.4 Public and Occupational Health and Safety

The assessments of potential incremental radiological and chemical impacts associated with this alternative are presented in this section. Summaries of radiological impacts from normal operations are presented in Tables 4–40 and 4–41 for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in Tables 4–42 and 4–43. The impacts from hazardous chemical releases during accident conditions are presented in Table 4–44. Background information on the effects of radiation on human health and safety is presented in Appendix E, Section E.2.

4.6.4.1 Normal Operations

Radiological Impacts

Under this alternative, radioactive releases would occur during sodium-bonded blanket spent nuclear fuel cleaning and driver spent nuclear fuel chopping and electrorefining. All of these activities are performed in the argon cell. Appendix E, Sections E.3, E.4.1, and E.4.2, details the method and assumptions used for calculating the impacts of normal operational radiological releases on the public health and safety. The maximum annual dose to the public would result when both blanket and driver spent nuclear fuel would be treated simultaneously under this alternative. Appendix E, Section E.4.2, provides details on the treatment process duration and throughputs for each fuel type.

Calculated maximum annual and project total radiological impacts to the public are given in **Table 4–40**. The impacts are calculated for two types of receptors: the general public living within 80 kilometers (50 miles) of

ANL-W in the year 2010, and a maximally exposed offsite individual (a member of the public assumed to be residing at the INEEL site boundary and receiving the maximum dose). Primary contributors to doses to members of the public are releases of tritium gas (about 1 percent of which was conservatively assumed to be in oxidized form) and krypton-85, which together contribute over 99.9 percent of the total calculated doses. To put the operational impacts into perspective, comparisons with impacts from natural background radiation also are included in the table. As shown in this table, the expected radiation doses to the maximally exposed offsite individual and the general public are much smaller than the limit of 10 millirem per year set by the EPA (40 CFR 61) and DOE (DOE Order 5400.5).

Table 4–40 Annual and Project Total Radiological Impacts to the Public From Operational Activities Under Alternative 4

Receptor	*Electrometallurgically Treat Driver Spent Nuclear Fuel*	*Clean and Melt and Dilute Blanket Spent Nuclear Fuel*	*Total*
Population Within 80 Kilometers (50 Miles) in the Year 2010			
Collective dose (person-rem per year) [a]	0.0027	0.00028	0.0030
Excess latent cancer fatalities (per year)	1.4×10^{-6}	1.4×10^{-7}	1.5×10^{-6}
Project total excess latent cancer fatalities [b]	8.0×10^{-6}	3.4×10^{-7}	8.3×10^{-6}
Maximally Exposed Offsite Individual			
Dose (millirem per year) [a]	0.00033	0.000048	0.00038
Percent of annual background radiation [c]	0.000092	0.000013	0.00011
Latent cancer fatality risk (per year)	1.7×10^{-10}	2.4×10^{-11}	1.9×10^{-10}
Project total lifetime cancer fatality risk [b]	9.4×10^{-10}	5.8×10^{-11}	1.0×10^{-9}
Average Individual Within 80 Kilometers (50 Miles)			
Dose (millirem per year) [d]	0.000011	1.2×10^{-6}	0.000012
Latent cancer fatality risk (per year)	5.6×10^{-12}	5.8×10^{-13}	6.2×10^{-12}
Project total lifetime cancer fatality risk [b]	3.3×10^{-11}	1.4×10^{-12}	3.5×10^{-11}

[a] Annual maximum dose during normal operations.
[b] Total calculated risk over 13 years. Majority of the dose is a result of 9 years of operation, electrometallurgically treating driver spent nuclear fuel, and cleaning of blanket spent nuclear fuel. Limited offsite consequences are associated with the melt and dilute processing of cleaned spent nuclear fuel.
[c] The annual natural background radiation level at INEEL is about 360 millirem for the average individual (see Table 3–8); the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 person-rem.
[d] Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of ANL-W in the year 2010 (240,338).

Table 4–41 summarizes worker population doses. Occupational doses were estimated by examining the type and duration of various operations performed by ANL-W workers involved with sodium-bonded spent nuclear fuel electrometallurgical and melt and dilute treatment processes. It was concluded that the average worker dose would not be different from what currently is being experienced. The estimated annual collective worker dose would be 22 person-rem, with an average individual dose of 60 millirem per year for each of the 346 involved workers. If these estimates were extended over the 13 years of treatment activities, and the dose (33 person-rem) from one year of deactivation activities was incorporated, the project total worker population dose would be 319 person-rem, leading to a risk of 0.13 latent cancer fatalities.

Table 4–41 Annual and Project Total Radiological Impacts to Workers From Operational Activities Under Alternative 4

Receptor	*Impacts*
Worker [a]	
Average worker dose (millirem per year)	60
Average worker latent cancer fatality risk (project total over 13 years)	0.00031
Worker Population	
Collective dose (person-rem per year) [b]	22
Excess latent cancer fatalities (per year)	0.0088
Project total dose (person-rem)	319
Project excess total latent cancer fatalities	0.13

[a] The regulatory dose limit for an individual worker is 5,000 millirem per year (10 CFR 835). However, the maximum annual dose to an involved worker would be kept below the DOE Administrative Control Level of 2,000 millirem per year, as established for all DOE activities in DOE Order N 441.1.

[b] The worker dose during one year of facility deactivation would be 33 person-rem.

Source: ANL 1999.

As shown in Tables 4–40 and 4–41:

- The annual dose to the maximally exposed offsite individual would be 0.00038 millirem per year, with an associated risk of developing a fatal cancer of 1.9×10^{-10} per year (or one chance in 5.3 billion that the individual would develop a fatal cancer per year of exposure).

- The collective dose to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 0.0030 person-rem per year, with an associated 1.5×10^{-6} latent cancer fatalities per year (or one chance in 670,000 that the population would experience a fatal cancer per year of exposure).

- The project total risk to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 8.3×10^{-6} latent cancer fatalities (or one chance in 120,000 that the exposed population would experience a fatal cancer).

- The collective dose to facility workers would be 22 person-rem per year, with an associated 0.0088 latent cancer fatalities per year (or one chance in 113 that the workers would experience a fatal cancer per year of operation).

- The project total dose to facility workers would be 319 person-rem, with an associated 0.13 latent cancer fatalities (or one chance in eight that the exposed workers would experience a fatal cancer).

These results indicate that no increase in the expected number of fatal cancers among the various affected populations would result from the activities under this alternative.

Hazardous Chemical Impacts

The hazardous chemical impacts to the public residing at the INEEL site boundary from the operational activities at ANL-W under Alternative 4 would be similar to the impacts evaluated for Alternative 1, as described in Section 4.3.4.1. The results indicate that no adverse toxic health or cancer effects would be expected from exposure to hazardous chemicals released under this alternative. The existing chemical environment is presented in Section 3.2.10.2.

4.6.4.2 Facility Accidents

Radiological Impacts

Potential radiological impacts to the public and a noninvolved onsite worker due to accidents during operational activities associated with cleaning (sodium removal) blanket spent nuclear fuel for melt and dilute processing and treating driver spent nuclear fuel using electrometallurgical treatment are summarized and presented in this section. The detailed analysis of facility accidents, with their associated assumptions, is presented in Appendix F. The detailed analysis considered a wide spectrum of potential accident scenarios, including fire, spills, criticality, earthquake, and aircraft crash. The aircraft crash and criticality events were determined to have an occurrence frequency of less than 10^{-7} per year and consequence analyses for these two events were not performed. Processing of blanket spent nuclear fuel would be performed in the Hot Fuel Examination Facility; treatment of driver spent nuclear fuel would be performed in both the Hot Fuel Examination Facility and the Fuel Conditioning Facility. Because driver spent nuclear fuel processing would take place in both of these facilities, the beyond-design-basis earthquake event is assessed for the driver spent nuclear fuel, taking into account the multifacility impacts of this event, and releases from both the Hot Fuel Examination Facility and the Fuel Conditioning Facility from the single earthquake event. The melt and dilute processing of blanket spent nuclear fuel would be performed only in the Hot Fuel Examination Facility. Melt and dilute processing of the fuel could result in a greater number of accidents to be considered (waste processing-related events) in the assessment of accidents involving blanket spent nuclear fuel at ANL-W than declad and clean operations. The multifacility impacts of the beyond-design-basis earthquake are not relevant to the blanket spent nuclear fuel melt and dilute processing which occurs in only one facility. Therefore, only the higher frequency design-basis earthquake event was analyzed. **Table 4–42** presents the frequencies and consequences of the postulated set of accidents to the maximally exposed offsite individual, the offsite population residing within 80 kilometers (50 miles) of the facility, and a noninvolved worker located 100 meters (330 feet) to 230 meters (755 feet) from the facility. The 230-meter (755-foot) distance is the ANL-W bus staging area, which leads to a higher dose to the noninvolved worker for the scenarios with elevated releases.

The dose to the maximally exposed offsite individual was calculated for the 95th percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50th percentile meteorological conditions. DOE did not quantitatively estimate the involved worker dose due to accidents (see the discussion on the involved worker in Section 4.2.4.2). The accident risks for the same receptors are summarized in **Table 4–43**.

For accidents at ANL-W, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be 1.9×10^{-6} per year (or one chance in 526,300 that the individual would develop a fatal cancer per year of operation) and 4.9×10^{-8} per year (or one chance in 20.4 million that the individual would develop a fatal cancer per year of operation), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.00022 per year (or one chance in 4,545 that the population would experience a fatal cancer per year of operation).

Hazardous Chemical Impacts

Nonradiological impacts are evaluated in terms of comparison to ERPG. ERPG values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects (see Appendix F, Section F.3.1.2, for details).

The nonradiological impacts of potential facility accidents (hazardous chemical) associated with the treatment of driver spent nuclear fuel using the electrometallurgical treatment process are summarized in **Table 4–44**.

Table 4–42 Accident Frequency and Consequences Under Alternative 4

Accident	Frequency (event per year)	Maximally Exposed Offsite Individual		Population Within 80 Kilometers (50 Miles)		Noninvolved Worker	
		Dose (millirem)	Latent Cancer Fatality Risk [a]	Dose (person-rem)	Excess Latent Cancer Fatalities [b]	Dose (millirem)	Latent Cancer Fatality Risk [a]
Driver Spent Nuclear Fuel							
Salt powder spill	0.01	0.00046	2.3×10^{-10}	0.000098	4.9×10^{-8}	4.7×10^{-7}	1.9×10^{-13}
Salt transfer drop	1.0×10^{-7}	0.19	9.5×10^{-8}	0.022	0.000011	0.073	2.9×10^{-8}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Cask drop	0.01	0.030	1.5×10^{-8}	0.0035	1.8×10^{-6}	0.00084	3.4×10^{-10}
Design-basis earthquake	0.008	12	6.0×10^{-6}	1.4	0.0007	4.7	1.9×10^{-6}
Beyond-design-basis earthquake	0.00001	22,000	0.022	2,500	1.3	370	0.00015
Blanket Spent Nuclear Fuel							
Cask drop	0.01	0.0024	1.2×10^{-9}	0.00028	1.4×10^{-7}	0.000049	2.0×10^{-11}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Sodium fire [c]	0.008	5.9	3.0×10^{-6}	0.69	0.00035	0.054	2.2×10^{-8}
Design-basis earthquake	0.008	471	0.00024	56.1	0.028	15.2	6.1×10^{-6}
Waste handling spill	0.0024	15	7.5×10^{-6}	1.8	0.00090	0.49	2.0×10^{-7}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.
[c] The frequency of this accident is the frequency of the facility design-basis earthquake initiating a cell fire.

Table 4–43 Annual Cancer Risks Due to Accidents at ANL-W Under Alternative 4

Accident	Maximally Exposed Offsite Individual [a]	Population Within 80 Kilometers (50 Miles) [b]	Noninvolved Worker [a]
Driver Spent Nuclear Fuel			
Salt powder spill	2.3×10^{-12}	4.9×10^{-10}	1.9×10^{-15}
Salt transfer drop	9.5×10^{-15}	1.1×10^{-12}	2.9×10^{-15}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Cask drop	1.5×10^{-10}	1.7×10^{-8}	3.4×10^{-12}
Design-basis earthquake	4.8×10^{-8}	5.6×10^{-6}	1.5×10^{-8}
Beyond-design-basis earthquake	2.2×10^{-7}	0.000013	1.5×10^{-9}
Blanket Spent Nuclear Fuel			
Cask drop	1.2×10^{-11}	1.4×10^{-9}	2.0×10^{-13}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Sodium fire	2.4×10^{-8}	2.8×10^{-6}	1.7×10^{-10}
Design-basis earthquake	1.9×10^{-6}	0.00022	4.9×10^{-8}
Waste handling spill	1.8×10^{-8}	2.2×10^{-6}	4.8×10^{-10}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

Table 4–44 Nonradiological Impacts of Accidents Under Alternative 4

Accident	*Frequency (Event Per Year)*	*Receptor Location*	*Exposure*
Uranium handling accident	0.01	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Design-basis earthquake	0.0002	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Sodium fire	0.00001	Noninvolved worker	Sodium: less than ERPG-1
		Maximally exposed offsite individual	Sodium: less than ERPG-1
Uranium fire	0.00001	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Beyond-design-basis earthquake	0.00001	Noninvolved worker	Cadmium: less than ERPG-1
			Uranium: less than ERPG-1
		Maximally exposed offsite individual	Cadmium: less than ERPG-1
			Uranium: less than ERPG-1

ERPG = Emergency Response Planning Guideline.

4.6.5 Environmental Justice

As discussed in Appendix H, Executive Order 12898 directs Federal agencies to address the disproportionately high and adverse health or environmental effects of alternatives on minority or low-income populations.

Analyses of normal operations and accident conditions presented in Section 4.6.4 show the risk of latent cancer fatalities to the public residing within 80 kilometers (50 miles) of the electrometallurgical and melt and dilute treatment processing facilities at ANL-W to be much lower than 1. Therefore, radiological and nonradiological risks posed by implementation of this alternative would not result in disproportionately high and adverse consequences to any particular group within the general population, including minority or low-income populations.

4.6.6 Waste Management

This alternative would generate process waste from treatment operations, other associated process waste from normal support operations, and deactivation waste following the conclusion of operations. Process waste would include high-level radioactive metallic and ceramic waste. Other associated process waste would include operational waste such as failed equipment, rags, packaging materials, and other miscellaneous items. Deactivation waste would include the disposal of process equipment and other materials. The fuel hardware in this alternative is used as additional steel in the melt and dilute process. All of these materials would be categorized according to existing DOE Orders and ANL-W waste management procedures. The anticipated categorization of the waste types generated and their expected interim storage and final disposal locations are given in Table 4–9 (see Section 4.2.6). The quantities of various waste forms generated as a result of Alternative 4 are provided in **Table 4–45**.

Estimates of the total amount of other associated process waste that would be generated are based on an evaluation of waste forecasts from ANL-W that accounted only for the fraction of total ANL-W waste that would be attributable to the treatment of sodium-bonded spent nuclear fuel. The values in Table 4–45 are for disposal and account for volume reduction. It is anticipated that a large fraction of the low-level radioactive waste that would be generated as a result of Alternative 4 could be volume-reduced at the Waste Experimental

Reduction Facility at INEEL prior to disposal at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex.

Table 4–45 Amounts of Waste Generated at ANL-W Under Alternative 4[a]

Waste Stream	Waste Quantities	
	Volume (cubic meters)	Mass (kilograms)
Direct Process Waste		
Fuel assembly hardware (low-level radioactive waste)	0	0
High-level radioactive ceramic waste	16.3 (26 canisters) [b]	24,400
High-level radioactive metallic waste	1.3 (2 canisters) [b]	2,500
Melt and dilute product	45.6 (114 canisters) [b]	114,000
Other Associated Process Waste		
High-level radioactive waste	0.4 (1 canister) [b]	220
Low-level radioactive waste [c]	650	132,000
Transuranic waste	11.2	4,730
Mixed waste	32.1	17,300
Sanitary waste	4,960	1.72×10^6
Deactivation Waste		
Low-level radioactive waste [c]	195	66,000
Transuranic waste	1.6	853
Mixed waste	5.6	3,600

[a] These waste generation estimates are through the year 2015. This is the assumed date that these materials might be sent to the repository. Treatment, high-level radioactive waste processing, deactivation, and interim storage would be accomplished during this time period.
[b] Standardized canisters.
[c] The volumes listed represent final disposal volumes following volume reduction at the Waste Experimental Reduction Facility at INEEL.
Source: ANL 1999.

The waste values in Table 4–45 are the total quantities that would be produced as a result of Alternative 4 operations. They are not incremental increases over the volumes provided in Table 4–10 that would result from the No Action Alternative. In Alternative 4, both the driver and blanket spent nuclear fuel would be transformed into high-level radioactive waste forms (ceramic and metallic, and a melt and dilute product) for disposal in the repository. In this conversion process, the total volume of material to be disposed of in the repository would be reduced from direct disposal values.

Direct Process Waste

For this alternative, fuel assembly hardware would be used as part of the required stainless steel to form the material ingot for disposal of the blanket spent nuclear fuel by melting. Its mass is included as part of the spent nuclear fuel disposal.

Under this alternative, metallic and ceramic high-level radioactive waste would be a primary product of the electrometallurgical treatment of driver spent nuclear fuel. The salt removed from the electrorefiners would contain the majority of fission products and transuranics from the spent nuclear fuel. This removed salt would be packaged and transferred to the Hot Fuel Examination Facility for processing into ceramic waste. The metallic waste form would consist primarily of stainless steel cladding hulls containing the noble metal fission products. The hulls would be removed from the electrorefiner and packaged for shipment to the Hot Fuel Examination Facility for processing into the metallic waste form. Both the ceramic and metallic waste would

be categorized as high-level radioactive waste. The volumes of waste forms provided in Table 4–45 are for the standardized canisters required for disposal of these materials.

The metallic and ceramic high-level radioactive waste and the melt and dilute product that would be generated at ANL-W would be stored temporarily at the Radioactive Scrap and Waste Facility at ANL-W to allow retrieval for future disposal. The Radioactive Scrap and Waste Facility was designed and constructed for temporary storage of this type of waste. Shielding would be provided by a combination of (1) steel storage liners which would store the waste, and (2) the soil surrounding the liners. When a geologic repository is available, the waste cans containing these materials would be removed from storage, shipped to the INEEL Dry Transfer Facility, and prepared for shipment to the repository.

Other Associated Process High-Level Radioactive Waste

High-level radioactive waste would be generated as a result of blanket spent nuclear fuel processing at ANL-W. This would result from activities in the Hot Fuel Examination Facility. Material in this waste stream would consist of the absorbent used in the off-gas system which has collected the volatile radionuclides released from the spent nuclear fuel when heated.

The volume of high-level radioactive waste that would be generated is expected to be less than the amount needed to fill a single standardized waste canister. Conservatively, the volume of a single canister, 0.4 cubic meters, has been used for the volume of high-level radioactive waste generated.

Other Associated Process Low-Level Radioactive Waste

Low-level radioactive waste would be generated as a result of processing sodium-bonded spent nuclear fuel at ANL-W. This would result from activities in the Fuel Conditioning Facility and the Hot Fuel Examination Facility (e.g., equipment decontamination and repair), as well as in other facilities at ANL-W (e.g., analytical laboratory activities). Material in this waste stream has been generated and routinely handled at ANL-W for many years.

The volume of low-level radioactive waste resulting from electrometallurgical and melt and dilute treatment processing of sodium-bonded spent nuclear fuel at ANL-W that would require disposal (after volume reduction) would be approximately 50 cubic meters (1,766 cubic feet) per year. This represents approximately 1 percent of the total annual volume of low-level radioactive waste currently being disposed of at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex, and the total 650 cubic meters (22,955 cubic feet) represent approximately 0.6 percent of the total Radioactive Waste Management Complex disposal capacity.

Other Associated Process Transuranic Waste

Transuranic waste would be generated from decontamination activities for repair and maintenance of items, and miscellaneous work associated with processing the sodium-bonded spent nuclear fuel. Transuranic waste would be generated primarily from activities conducted in gloveboxes and hot cells at ANL-W.

All of the transuranic waste that would be generated at ANL-W would be packaged and certified in accordance with Waste Isolation Pilot Plant Acceptance Criteria prior to transport to the Waste Isolation Pilot Plant. The transuranic waste generated would amount to approximately 1 cubic meter (35 cubic feet) per year, which would be approximately 0.002 percent of the volume of transuranic waste in retrievable storage at the Radioactive Waste Management Complex at INEEL. The total volume of transuranic waste would be 11.2 cubic meters (395 cubic feet), which would be less than 0.006 percent of the estimated total volume of transuranic waste to be placed at the Waste Isolation Pilot Plant.

Other Associated Process Mixed Waste

Mixed waste would be generated primarily from the disposal of any cadmium-contaminated equipment or cleanup material and the analysis of cadmium samples. Mixed waste would be handled according to ANL-W procedures that require limited accumulation at the point of generation. Interim storage of this waste would be accomplished at the Radioactive Scrap and Waste Facility prior to eventual disposal. The Radioactive Scrap and Waste Facility is a permitted mixed waste storage facility for these materials. The mixed waste streams that contribute to the overall mixed waste generated by electrometallurgical treatment have been identified in the INEEL Site Treatment Plan (DOE 1995b).

Deactivation Waste

A variety of waste would be generated as part of deactivation activities associated with processing at ANL-W. This waste would include process equipment and process material, such as electrorefiner cadmium from electrometallurgical treatment of driver spent nuclear fuel. Generated waste categories would include low-level radioactive waste, transuranic waste, and mixed waste. This waste would be categorized and disposed of according to DOE Orders and ANL-W radioactive waste management procedures, as described above for each waste category.

The largest volume of deactivation waste would be low-level radioactive waste, generated as a result of dismantling and disposal (electrometallurgical treatment and melt and dilute equipment). Components of electrometallurgical treatment of the driver spent nuclear fuel that would require disposition include two electrorefiners, two hot hydrostatic presses, and two V-mixers, as well as other components such as the grinder/crusher. Decontamination of these components would generate additional mixed, transuranic, and low-level radioactive waste that would require management. The deactivation waste volume would be generated in a single year. This waste would represent an increase of approximately three times the annual waste generated by the treatment operations of Alternative 4. The total deactivation waste would represent an additional 30 percent over the total associated process waste requiring disposal.

4.7 ALTERNATIVE 5: DECLAD AND CLEAN BLANKET FUEL AND ELECTROMETALLURGICALLY TREAT DRIVER FUEL AT ANL-W; MELT AND DILUTE BLANKET FUEL AT SRS

Under this alternative, the sodium-bonded blanket spent nuclear fuel would be declad and cleaned to remove metallic sodium at ANL-W, packaged in aluminum cans, and shipped to SRS for treatment using the melt and dilute process at Building 105-L. The melt and dilute product from the treatment process would be stored at SRS pending repackaging and transportation for disposal in a geologic repository. The removed sodium would be stabilized using an oxidation/carbonation process (ANL 1999). The sodium-bonded driver spent nuclear fuel would be treated at ANL-W using the electrometallurgical treatment process. The process steps for the electrometallurgical treatment of driver spent nuclear fuel would be similar to those described earlier in Section 4.3 and in Appendix C. The treatment of driver spent nuclear fuel using the electrometallurgical process could start as early as 2000 and could be completed by 2006 to 2007. The preparation of blanket spent nuclear fuel and its shipment to SRS could start in 2003 and could be completed by 2009. In addition, a full year of operations would be needed to deactivate the processing equipment and establish an industrially safe configuration.

Current planning at SRS has scheduled the melt and dilute process at Building 105-L for other missions until 2035 (DOE 2000). Melt and dilute process of blanket spent nuclear fuel at SRS could start around 2020, if capacity becomes available, and could be completed by 2023.

4.7.1 Air Quality

Nonradiological Gaseous Emissions

It is expected that criteria and hazardous air pollutants released from operational activities at ANL-W under this alternative would be the same as for Alternative 1, as described in Section 4.3.1 (see also Appendix E, Section E.5.3.1, for more detail). Baseline air quality concentrations are presented in Section 3.2.3.1.

At SRS, nonradiological air emissions would result from operation of ancillary support facilities for the melt and dilute process at Building 105-L. The concentrations of nonradiological air pollutants attributed to this alternative at SRS are presented in **Table 4–46**. These concentrations are based on information in the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000) for the melt and dilute processing of similar fuel (see Appendix E, Section E.5.3.2, for more details). The site boundary concentrations are the incremental concentrations that would be generated in this alternative plus the baseline concentrations given in Section 3.3.3.1. Only those air pollutants that are expected to be emitted under this alternative and have ambient air quality standards are presented in the table. Note that SRS has no Prevention of Significant Deterioration increment-consuming sources on site; therefore, a Prevention of Significant Deterioration increment analysis was not performed. SRS is located in an area of attainment for criteria pollutants; therefore, a confirmatory analysis is not required for this alternative. Health effects from hazardous chemicals associated with this alternative are addressed in Section 4.7.4.1.

Radiological Gaseous Emissions

The decladding and cleaning of the blanket spent nuclear fuel and the electrometallurgical treatment of the driver spent nuclear fuel at ANL-W would release gaseous fission products to the argon cell environment. Krypton-85 and elemental tritium would be the most prevalent radioactive gaseous fission products released to the environment. The tritium released in the cell would not be oxidized due to a very low presence of oxygen and humidity in the argon cell. The argon cell also would contain an equilibrium concentration of other radionuclide isotopes. Appendix E, Section E.4.1, provides a list of various isotopes that would be present in the argon cell in nanocuries (10^{-9} curies) and would be released to the atmosphere through the facility stack along with krypton-85 and elemental tritium. The maximum release of radioactive gaseous emissions would occur when decladding the blanket spent nuclear fuel for packaging and shipment to SRS and electrometallurgical treatment of driver spent nuclear fuel are performed simultaneously. It was estimated that this simultaneous operation would occur over a three-year period starting in 2003 (see Appendix E, Section E.4.2, for details on releases during the processing period at ANL-W under this alternative). Based on an annual decladding throughput of 10 metric tons of heavy metal of blanket spent nuclear fuel and an electrometallurgical treatment process of about 0.6 metric tons of heavy metal of driver spent nuclear fuel, about 809 curies of elemental tritium and 11,860 curies of gaseous krypton-85 would be released annually to the atmosphere.

Since declad and cleaned fuel would be packaged and sent to SRS, some gaseous fission products would be expected in that fuel. However, it was conservatively assumed that the gaseous fission products in the blanket spent nuclear fuel also would be released to the environment during the melt and dilute process at SRS. The radiological exposures of the public and workers from radioactive emissions are presented in Section 4.7.4.

4.7.2 Water Resources

As stated in Section 4.4.2, the decladding and cleaning of blanket spent nuclear fuel and electrometallurgical treatment of driver spent nuclear fuel would not discharge any radiological chemical material to the surface or groundwater at the INEEL site. These activities also would not impact the current groundwater usage at the site. For a discussion of impacts on water resources at ANL-W; see Section 4.4.2.

Table 4–46 Nonradiological Air Quality Concentrations at the Site Boundary Under Alternative 5 at SRS for Comparison With Ambient Air Quality Standards

Pollutant	*Averaging Period*	*Most Stringent Standard or Guideline (micrograms per cubic meter)*	*Maximum Incremental Concentration (micrograms per cubic meter)*
Criteria Pollutants			
Carbon monoxide	8 hours 1 hour	10,000 40,000	0.08 0.51
Nitrogen dioxide	Annual	100	Less than 0.01
PM_{10}	Annual 24 hours (interim) 24 hours (99th percentile over 3 years)	50 150 150	Not detectable Not detectable Not available
$PM_{2.5}$	3-year annual 24 hours (98th percentile over 3 years)	15 65	Not available Not available
Sulfur dioxide	Annual 24 hours 3 hours	80 365 1,300	0.01 0.03 Not detectable
State-regulated Pollutants			
Gaseous fluoride	30 days 7 days 24 hours 12 hours	0.8 1.6 2.9 3.7	Not detectable Not detectable Not detectable Not detectable
Total suspended particulates	Annual	75	Less than 0.01
Hazardous/Toxic Compounds			
1,1,1-trichloroethane	24 hours	9,550	Less than 0.01
Benzene	24 hours	150	Not detectable
Ethanolamine	24 hours	200	Less than 0.01
Ethyl benzene	24 hours	4,350	Not detectable
Ethylene glycol	24 hours	650	Less than 0.01
Formaldehyde	24 hours	15	Less than 0.01
Glycol ethers	24 hours	No standard	Less than 0.01
Hexachloronaphthalene	24 hours	1	Less than 0.01
Hexane	24 hours	900	Less than 0.01
Manganese	24 hours	25	Not detectable
Methyl alcohol	24 hours	1,310	Less than 0.01
Methyl-ethyl-ketone	24 hours	14,750	Less than 0.01
Methyl-isobutyl-ketone	24 hours	2,050	Not detectable
Methylene chloride	24 hours	8,750	Not detectable
Naphthalene	24 hours	1,250	Less than 0.01
Nitric acid	24 hours	125	Not detectable
Phenol	24 hours	190	Not detectable
Phosphorous	24 hours	0.5	Not detectable
Sodium hydroxide	24 hours	50	Not detectable
Toluene	24 hours	2,000	Less than 0.01
Trichloroethane	24 hours	6,750	Not detectable
Vinyl acetate	24 hours	176	Not detectable
Xylene	24 hours	4,350	Less than 0.01

PM_n = Particulate matter less than or equal to *n* microns in diameter.
Source: Bickford et al. 1997.

The impacts on water resources from treating blanket spent nuclear fuel at Building 105-L using the melt and dilute process are described below.

Surface Water

No surface water would be used for the melt and dilute processing of blanket spent nuclear fuel at Building 105-L. Building 105-L is outside the 100-year floodplain, as shown in Figure 3-6.

Nonradiological Liquid Effluent

No nonradiological liquid effluent would be generated by melting and diluting blanket spent nuclear fuel at Building 105-L. Sanitary waste would be treated at the SRS Central Wastewater Treatment Facility and discharged through an existing NPDES-permitted outfall. Since employment would not increase as a result of processing this fuel, the treatment rates through the Central Wastewater Treatment Facility would not be affected and the requirements of the SRS NPDES permit would continue to be met (DOE 2000).

Radiological Liquid Effluent

No radiological liquid effluent would be discharged to surface water from the melt and dilute process at Building 105-L.

Groundwater

Process water would not be required for the melt and dilute process at Building 105-L. Domestic water would come from groundwater. No increase in domestic water use is anticipated since no increase in employment is expected to result from the melt and dilute operation.

Nonradiological Liquid Effluent

No nonradiological chemicals would be discharged to groundwater from the melt and dilute processing at Building 105-L.

Radiological Liquid Effluent

No radiological liquid effluent would be discharged to groundwater from the melt and dilute process at Building 105-L.

4.7.3 Socioeconomics

Under this alternative, the existing facilities at ANL-W and SRS would remain operational. No new employment or in-migration of workers would be required. Thus, there would be no additional impacts on the socioeconomic conditions in the region around INEEL and SRS.

4.7.4 Public and Occupational Health and Safety

The assessments of potential incremental radiological and chemical impacts associated with this alternative are presented in this section. Summaries of radiological and chemical impacts from normal operations are presented in Tables 4–47 through 4–48 for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in Tables 4–50 through 4–53. The impacts from hazardous chemical releases during accident conditions are similar to those presented in Section 4.5.4.1.

Background information on the effects of radiation on human health and safety is presented in Appendix E, Section E.2.

4.7.4.1 Normal Operations

Radiological Impacts

Under this alternative, radioactive releases would occur during sodium-bonded blanket spent nuclear fuel decladding and cleaning, driver spent nuclear fuel chopping, and electrorefining. All of these activities would be performed in the argon cell. Appendix E, Sections E.3, E.4.1, and E.4.2, details the method and assumptions used for calculating the impacts of normal operational radiological releases on the public health and safety. The maximum annual dose to the public at ANL-W would result when decladding and cleaning of blanket spent nuclear fuel and treatment of driver spent nuclear fuel are performed simultaneously under this alternative. The doses from decladding and cleaning blanket spent nuclear fuel and treating driver spent nuclear fuel at ANL-W would be similar to those presented for Alternative 2 in Section 4.4.4.1.

Calculated maximum annual and projected total radiological impacts to the public are given in **Table 4–47**. The impacts are calculated for two types of receptors: the general public living within 80 kilometers (50 miles) of ANL-W and Building 105-L at SRS in the year 2010, and a maximally exposed offsite individual (a member of the public assumed to be residing at the INEEL or SRS site boundary and receiving the maximum dose). Primary contributors to doses to members of the public at ANL-W would be from releases of tritium gas (about 1 percent of which were assumed conservatively to be in oxidized form) and krypton-85, which together would contribute over 99.9 percent of the total calculated doses. As shown in this table, the expected radiation doses to the maximally exposed offsite individual and the general public would be much smaller than the limit of 10 millirem per year set by the EPA (40 CFR 61) and DOE (DOE Order 5400.5).

The blanket spent nuclear fuel would be declad and cleaned at ANL-W, where it is expected that the gaseous fission products would be released. However, for the melt and dilute process, it is assumed conservatively that these gaseous fission products would be released at SRS. The melt and dilute process is assumed to continue for three years. (Appendix E, Section E.4.4, provides the details on the treatment process duration.) To put the operational impacts into perspective, comparisons with impacts from natural background radiation also are included in the table.

Table 4–48 summarizes worker population doses. Occupational doses were estimated by examining the type and duration of various operations performed by SRS workers involved with the melt and dilute process. The estimated annual collective worker dose would be 50 person-rem, with an average individual dose of 500 millirem per year for each of the 100 involved workers. If these estimates were projected for maximum process activities over three years, the project total worker population dose would be 150 person-rem, leading to a risk of 0.06 latent cancer fatalities. The estimated annual collective worker dose from decladding and cleaning blanket spent nuclear fuel and electrometallurgically treating driver spent nuclear fuel at ANL-W is 22 person rem, as indicated in Section 4.4.4.1.

As shown in Tables 4–47 and 4–48:

- The annual dose to the maximally exposed offsite individual at ANL-W would be 0.00038 millirem per year, with an associated risk of developing a fatal cancer of 1.9×10^{-10} per year (or one chance in 5.3 billion that the individual would develop a fatal cancer per year of exposure).

Table 4–47 Annual and Project Total Radiological Impacts to the Public From Operational Activities Under Alternative 5

Receptor	*Melt and Dilute Blanket Spent Nuclear Fuel at SRS*	*Clean Blanket Spent Nuclear Fuel and Electrometallurgically Treat Driver Spent Nuclear Fuel at ANL-W*
Population Within 80 Kilometers (50 Miles) in the Year 2010		
Collective dose (person-rem per year) [a]	0.0076	0.0030
Excess latent cancer fatalities (per year)	3.8×10^{-6}	1.5×10^{-6}
Project total excess latent cancer fatalities [b]	0.000011	8.3×10^{-6}
Maximally Exposed Offsite Individual		
Dose (millirem per year) [a]	0.00010	0.00038
Percent of annual background radiation [c]	0.000033	0.00011
Latent cancer fatality risk (per year)	5.0×10^{-11}	1.9×10^{-10}
Project total lifetime cancer fatality risk	1.5×10^{-10}	1.0×10^{-9}
Average Individual Within 80 Kilometers (50 Miles)		
Dose (millirem per year) [d]	0.000011	0.000012
Latent cancer fatality risk (per year)	5.5×10^{-12}	6.2×10^{-12}
Project total lifetime cancer fatality risk [b]	1.6×10^{-11}	3.5×10^{-11}

[a] Annual maximum dose during normal operations.
[b] Total calculated risk over nine years at ANL-W and three years at SRS.
[c] The annual natural background radiation level at INEEL and at SRS is about 360 and 300 millirem, respectively, for the average individual (see Tables 3–8 and 3–20); the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 person-rem at INEEL and 254,000 person-rem at SRS.
[d] Obtained by dividing the population dose by the number of people projected to live in the year 2010 within 80 kilometers (50 miles) of ANL-W (240,338) and SRS Building 105-L (715,000).

Table 4–48 Annual and Project Total Radiological Impacts to Workers From Operational Activities Under Alternative 5

Receptor	*Impacts*	
Worker [a]	Operations at SRS	Operations at ANL-W
Average worker dose (millirem per year)	500	60
Average worker latent cancer fatality risk (project total)	0.00060 [b]	0.00022 [c]
Worker Population		
Collective dose (person-rem per year)	50	22
Excess latent cancer fatalities (per year)	0.020	0.0088
Project total dose (person-rem)	150	231
Project total excess latent cancer fatalities	0.06 [b]	0.092

[a] The regulatory dose limit for an individual worker is 5,000 millirem per year (10 CFR 835). However, the maximum annual dose to a worker would be kept below the DOE Administrative Control Level of 2,000 millirem per year, as established for all DOE activities in DOE Order N 441.1.
[b] Operations at SRS to treat blanket spent nuclear fuel using melt and dilute processing at Building 105-L would be performed over three years.
[c] Operations at ANL-W to declad and clean blanket spent nuclear fuel and treat driver spent nuclear fuel would be performed over nine years. The project total dose includes 33 person-rem from one year of facility deactivation activities.

Sources: ANL 1999, DOE 2000.

- The collective dose to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 0.0030 person-rem per year, with an associated 1.5×10^{-6} latent cancer fatalities per year (or one chance in 670,000 that the population would experience a fatal cancer per year of exposure).

- The project total risk to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 8.3×10^{-6} latent cancer fatalities (or one chance in 120,000 that the exposed population would experience a fatal cancer).

- The collective dose to ANL-W facility workers would be 22 person-rem per year, with an associated 0.0088 latent cancer fatalities per year (or one chance in 113 that the workers would experience a fatal cancer per year of operation).

- The project total dose to ANL-W facility workers would be 231 person-rem, with an associated 0.092 latent cancer fatalities (or one chance in 11 that the exposed workers would experience a fatal cancer).

- The annual dose to the maximally exposed offsite individual from melt and dilute processing at Building 105-L would be 0.00010 millirem per year, with an associated risk of developing a fatal cancer of 5×10^{-11} per year (or one chance in 20 billion that the individual would develop a fatal cancer per year of exposure).

- The collective dose to the population within 80 kilometers (50 miles) of Building 105-L would be 0.0076 person-rem per year, with an associated 3.8×10^{-6} latent cancer fatalities per year (or one chance in 263,100 that the population would experience a fatal cancer per year of exposure).

- The project total risk to the population within 80 kilometers (50 miles) of Building 105-L would be 0.000011 latent cancer fatalities (or one chance in 91,000 that the exposed population would experience a fatal cancer).

- The collective dose to Building 105-L facility workers would be 50 person-rem per year, with an associated 0.020 latent cancer fatalities (or one chance in 50 that the workers would experience a fatal cancer per year of operation).

- The project total dose to Building 105-L facility workers would be 150 person-rem, with an associated 0.06 latent cancer fatalities (or one chance in 17 that the exposed workers would experience a fatal cancer).

These results indicate that no increase in the expected number of fatal cancers among the various affected populations would result from the activities under this alternative.

Hazardous Chemical Impacts

The hazardous chemical impacts to the public residing at the INEEL site boundary from the operational activities at ANL-W under Alternative 5 would be similar to the impacts evaluated for Alternative 1, as described in Section 4.3.4.1. The results indicate that no adverse toxic health or cancer effects would be expected from exposure to hazardous chemicals released under this alternative. The existing baseline chemical environment is presented in Section 3.2.10.2.

For SRS, both carcinogenic and noncarcinogenic health effects to the public were assessed from exposure to hazardous chemicals, and the results are summarized in **Table 4–49**. The results indicate that no adverse toxic health or cancer effects would be expected from exposure to hazardous chemicals released at SRS under this alternative (see Appendix E, Section E.5.3, for more details). The existing baseline chemical environment is presented in Section 3.3.10.2.

Table 4–49 Hazardous Chemical Impacts to the Public From Operational Activities at SRS Under Alternative 5

Chemical	*Annual Concentration (milligrams per cubic meter)*	*Hazard Quotient (noncarcinogenic chemicals)*	*Cancer Risk (carcinogenic chemicals)*
Formaldehyde	1.3×10^{-6}	None	1.6×10^{-8}
Hexane	1.3×10^{-6}	6.3×10^{-6}	None
Manganese	Not detectable	Not detectable	None
Methyl ethyl ketone	1.3×10^{-6}	1.3×10^{-6}	None
Naphthalene	1.3×10^{-6}	0.00042	None
Toluene	1.3×10^{-6}	3.1×10^{-6}	None
Hazard Index		0.00043	Not applicable

4.7.4.2 Facility Accidents

Radiological Impacts

The potential radiological impacts to the public and a noninvolved onsite worker resulting from accidents during decladding and cleaning and melting and diluting the blanket spent nuclear fuel elements, and from electrometallurgical treatment of driver spent nuclear fuel operational activities at ANL-W and SRS, are summarized and presented in this section. The detailed analysis of facility accidents, with associated assumptions, is presented in Appendix F. The detailed analysis considered a wide spectrum of potential accident scenarios, including fire, spills, criticality, earthquake, and aircraft crash. The aircraft crash and criticality events were determined to have an occurrence frequency of less than 10^{-7} per year, and consequence analyses for these two events were not performed. Processing of the blanket spent nuclear fuel would be performed in the Hot Fuel Examination Facility; treatment of the driver spent nuclear fuel is performed in both the Hot Fuel Examination Facility and the Fuel Conditioning Facility. Because the processing of the driver spent nuclear fuel would take place in both of these facilities, the beyond-design-basis earthquake event is assessed for the driver spent nuclear fuel, taking into account the multi-facility impacts of this event. The decladding and cleaning of the blanket spent nuclear fuel would be performed only in the Hot Fuel Examination Facility. The multi-facility impacts of the beyond-design-basis earthquake are not relevant to this blanket spent nuclear fuel processing. Therefore, only the higher frequency design-basis earthquake event was analyzed. **Table 4–50** presents the frequencies and consequences of the postulated set of accidents to the maximally exposed offsite individual; the offsite population residing within 80 kilometers (50 miles) of the facility; and a noninvolved worker located 100 meters (330 feet) to 230 meters (755 foot) from the facility. The 230-meter (755-foot) distance is the ANL-W bus staging area, which leads to a higher dose to the noninvolved worker for the scenarios with elevated releases.

The dose to the maximally exposed offsite individual was calculated for the 95th percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50th percentile meteorological conditions. DOE did not quantitatively estimate the involved worker dose due to accidents (see the discussion on the involved worker in Section 4.2.4.2). The accident risks for the same receptors are summarized in **Table 4–51**.

Table 4–50 Accident Frequency and Consequences at ANL-W Under Alternative 5

Accident	Frequency (event per year)	Maximally Exposed Offsite Individual		Population Within 80 Kilometers (50 Miles)		Noninvolved Worker	
		Dose (millirem)	Latent Cancer Fatality Risk [a]	Dose (person-rem)	Excess Latent Cancer Fatalities [b]	Dose (millirem)	Latent Cancer Fatality Risk [a]
Driver Spent Nuclear Fuel							
Salt powder spill	0.01	0.00046	2.3×10^{-10}	0.000098	4.9×10^{-8}	4.7×10^{-7}	1.9×10^{-13}
Salt transfer drop	1.0×10^{-7}	0.19	9.5×10^{-8}	0.022	0.000011	0.073	2.9×10^{-8}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Cask drop	0.01	0.030	1.5×10^{-8}	0.0035	1.8×10^{-6}	0.00084	3.4×10^{-10}
Design-basis earthquake	0.008	12	6.0×10^{-6}	1.4	0.0007	4.7	1.9×10^{-6}
Beyond-design-basis earthquake	0.00001	22,000	0.022	2,500	1.3	370	0.00015
Blanket Spent Nuclear Fuel							
Cask drop	0.01	0.0024	1.2×10^{-9}	0.00028	1.4×10^{-7}	0.000049	2.0×10^{-11}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Sodium fire [c]	0.008	5.9	3.0×10^{-6}	0.69	0.00035	0.054	2.2×10^{-8}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.
[c] The frequency for this accident is the frequency for the facility design-basis earthquake-initiating cell fire.

Table 4–51 Annual Cancer Risks Due to Accidents at ANL-W Under Alternative 5

Accident	Maximally Exposed Offsite Individual [a]	Population Within 80 Kilometers (50 Miles) [b]	Noninvolved Worker [a]
Driver Spent Nuclear Fuel			
Salt powder spill	2.3×10^{-12}	4.9×10^{-10}	1.9×10^{-15}
Salt transfer drop	9.5×10^{-15}	1.1×10^{-12}	2.9×10^{-15}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Cask drop	1.5×10^{-10}	1.7×10^{-8}	3.4×10^{-12}
Design-basis earthquake	4.8×10^{-8}	5.6×10^{-6}	1.5×10^{-8}
Beyond-design-basis earthquake	2.2×10^{-7}	0.000013	1.5×10^{-9}
Blanket Spent Nuclear Fuel			
Cask drop	1.2×10^{-11}	1.4×10^{-9}	2.0×10^{-13}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Sodium fire	2.4×10^{-8}	2.8×10^{-6}	1.7×10^{-10}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

For accidents at ANL-W, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be 2.2×10^{-7} per year (or one chance in 4.5 million that the individual would develop a fatal cancer per year of operation) and 1.5×10^{-8} per year (or one chance in 66.7 million that the individual would develop a fatal cancer per year of operation), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.000013 per year (or one chance in 76,920 that the population would experience a fatal cancer per year of operation).

The potential radiological impacts to the public and a noninvolved onsite worker due to accidents during melt and dilute operational activities at SRS are summarized below. The detailed analysis of facility accidents, with the associated assumptions, is presented in Appendix F. **Table 4–52** presents the frequencies and consequences of the postulated set of accidents to the maximally exposed offsite individual, the offsite population residing within 80 kilometers (50 miles) of the facility, and a noninvolved worker located 100 meters (330 feet) to 300 meters (980 feet) from the facility. The 300-meter (980-foot) distance leads to a higher dose to the noninvolved worker for the scenarios with elevated releases.

Table 4–52 Accident Frequency and Consequences at SRS Under Alternative 5

Accident	Frequency (event per year)	Maximally Exposed Offsite Individual		Population Within 80 Kilometers (50 Miles)		Noninvolved Worker	
		Dose (millirem)	Latent Cancer Fatality Risk [a]	Dose (person-rem)	Excess Latent Cancer Fatalities [b]	Dose (millirem)	Latent Cancer Fatality Risk [a]
Waste handling spill	0.0064	2.1	1.1×10^{-6}	3.6	0.0018	0.17	6.8×10^{-8}
Loss of power	0.006	2100	0.0011	3500	1.8	140	0.000056
Melter eruption/explosion [c]	0.0005	269	0.00014	1160	0.58	72.9	0.000029
Fire	0.075	86	0.000043	140	0.07	6.3	2.5×10^{-6}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.
[c] In the draft EIS, this accident was identified as "loss of cooling water." Consistent with the SRS Spent Nuclear Fuel Management Final EIS, the accident name was changed.

The dose to the maximally exposed offsite individual was calculated for the 95th percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50th percentile meteorological conditions. DOE did not quantitatively estimate the facility worker population dose due to accidents. The accident risks for the same receptors are summarized in **Table 4–53**.

Table 4–53 Annual Cancer Risks Due to Accidents at SRS Under Alternative 5

Accident	Maximally Exposed Offsite Individual [a]	Population Within 80 Kilometers (50 Miles) [b]	Noninvolved Worker [a]
Waste handling spill	6.7×10^{-9}	0.000012	5.5×10^{-10}
Loss of power	6.6×10^{-6}	0.011	3.4×10^{-7}
Melter eruption/explosion	7.0×10^{-8}	0.00029	1.5×10^{-8}
Fire	3.2×10^{-6}	0.0053	1.9×10^{-7}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

For accidents at SRS, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be 6.6×10^{-6} per year (or one chance in 151,500 that the individual would develop a fatal cancer per year of operation) and 3.4×10^{-7} per year (or one chance in 2.9 million that the individual would develop a fatal cancer per year of operation), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.011 per year (or one chance in 91 that the population would experience a fatal cancer per year of operation).

Hazardous Chemical Impacts

The impacts of accidents involving hazardous chemicals for this alternative are the same as those described in Section 4.5.4.2 for Alternative 3: Declad and clean blanket fuel and electrometallurgically treat driver fuel at ANL-W; PUREX process blanket fuel at SRS.

4.7.5 Environmental Justice

As discussed in Appendix H, Executive Order 12898 directs Federal agencies to address disproportionately high and adverse health or environmental effects of alternatives on minority or low-income populations.

Analyses of normal operations and accident conditions presented in Section 4.7.4 show the risk of latent cancer fatalities to the public residing within 80 kilometers (50 miles) of the electrometallurgical treatment and decladding and cleaning processing facilities at ANL-W and of the melt and dilute treatment facility at SRS to be much lower than 1. Therefore, radiological and nonradiological risks posed by implementation of this alternative would have no disproportionately high and adverse consequences on any particular group within the general population, including minority or low-income populations.

4.7.6 Waste Management

ANL-W

This alternative would generate process waste from treatment operations, other associated process waste from normal support operations, and deactivation waste following the conclusion of operations. Process waste would include fuel hardware and high-level radioactive metallic and ceramic waste. Other associated process waste would include operational waste such as failed equipment, rags, packaging materials, and other miscellaneous items. Deactivation waste would include the disposal of process equipment and other materials. All of these materials would be categorized according to existing DOE Orders and ANL-W waste management procedures. The anticipated categorization of these waste types generated at ANL-W and their expected interim storage and final disposal locations are given in Table 4–9 (see Section 4.2.6). The quantities of various waste forms generated as a result of Alternative 5 at ANL-W are provided in **Table 4–54.**

Estimates of the total amount of other associated process waste that would be generated are based on an evaluation of waste forecasts from ANL-W that accounted only for the fraction of total ANL-W waste that would be attributable to the treatment of sodium-bonded spent nuclear fuel. The values in Table 4–54 are for disposal and account for volume reduction. It is anticipated that a large fraction of the low-level radioactive waste that would be generated as a result of Alternative 5 could be volume-reduced at the Waste Experimental Reduction Facility at INEEL prior to disposal at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex.

The waste values in Table 4–54 are total quantities that would be produced as a result of Alternative 5 operations. They are not incremental increases over the volumes provided in Table 4–10 that would result from the No Action Alternative. In Alternative 5, the driver spent nuclear fuel would be transformed into high-level radioactive waste forms (ceramic and metallic) at ANL-W for disposal in the repository, and in this conversion process, the total volume of material to be disposed of in the repository would be reduced from direct disposal values. The blanket spent nuclear fuel would be cleaned and declad and sent to SRS for melt and dilute processing. The high-level radioactive waste (melt and dilute product) that would be generated from melt and dilute processing at SRS is presented in Table 4–56.

Table 4–54 Amounts of Waste Generated at ANL-W Under Alternative 5[a]

Waste Stream	Waste Quantities	
	Volume (cubic meters)	Mass (kilograms)
Direct Process Waste		
Fuel assembly hardware (low-level radioactive waste)	37.5	13,100
High-level radioactive ceramic waste	16.3 (26 canisters)[b]	24,400
High-level radioactive metallic waste	1.3 (2 canisters)[b]	2,500
Spent nuclear fuel	0	0
Other Associated Process Waste		
High-level radioactive waste	0.4 (1 canister)[b]	220
Low-level radioactive waste[c]	555	113,000
Transuranic waste	9.1	3,800
Mixed waste	27.5	14,800
Sanitary waste	4,960	1.7×10^6
Deactivation Waste		
Low-level radioactive waste[c]	178	60,000
Transuranic waste	1.6	853
Mixed waste	5.1	3,400

[a] These waste generation estimates are through the year 2015. This is the assumed date that these materials might be sent to the repository. Treatment, high-level radioactive waste processing, deactivation, and interim storage would be accomplished during this time period.

[b] Standardized canisters.

[c] The volumes listed represent final disposal volumes following volume reduction at the Waste Experimental Reduction Facility at INEEL.

Source: ANL 1999.

Direct Process Waste

For this alternative, fuel hardware would be removed from the fuel elements in the Fuel Conditioning Facility air cell and disposed of as low-level radioactive waste. These hardware components would be primarily stainless steel materials that contain short-lived radionuclides. This waste stream has been produced at ANL-W for many years and would be handled, as in the past, according to DOE Orders and ANL-W waste management procedures. In addition, the blanket spent nuclear fuel cladding would be included in the fuel hardware stream.

Under this alternative, metallic and ceramic high-level radioactive waste would be a primary product of the electrometallurgical treatment of driver spent nuclear fuel. The salt removed from the electrorefiners would contain the majority of fission products and transuranics from the spent nuclear fuel. This removed salt would be packaged and transferred to the Hot Fuel Examination Facility for processing into ceramic waste. The metallic waste form would consist primarily of stainless steel cladding hulls containing the noble metal fission products. The hulls would be removed from the electrorefiner and packaged for shipment to the Hot Fuel Examination Facility for processing into the metallic waste form. Both of these waste types would be categorized as high-level radioactive waste. The volumes of waste forms provided in Table 4–54 are for the standardized canisters required for disposal of these materials.

The metallic and ceramic high-level radioactive waste generated as a result of electrometallurgical treatment of driver spent nuclear fuel under this alternative, and the driver and blanket spent nuclear fuel from the demonstration project at ANL-W, would be stored temporarily at the Radioactive Scrap and Waste Facility at ANL-W to allow retrieval for future disposal. The Radioactive Scrap and Waste Facility was designed and

constructed for temporary storage of this type of waste. Shielding would be provided by a combination of (1) steel storage liners which would store the waste, and (2) the soil surrounding the liners. When a geologic repository is available, the waste cans containing the metallic and ceramic high-level radioactive waste would be removed from storage, shipped to the INEEL Dry Transfer Facility, and prepared for shipment to the repository.

Other Associated Process High-Level Radioactive Waste

High-level radioactive waste could be generated as a result of blanket spent nuclear fuel processing at ANL-W and SRS. This would result from activities in the Hot Fuel Examination Facility (at ANL-W) and Building 105-L (at SRS). Material in this waste stream would consist of the absorbent used in the off-gas system which has collected the volatile radionuclides released from the spent nuclear fuel when heated.

The volume of high-level radioactive waste that would be generated is expected to be less than the amount needed to fill a single standardized waste canister. Conservatively, the volume of a single canister, 0.4 cubic meters, has been used for the volume of high-level radioactive waste generated.

Other Associated Process Low-Level Radioactive Waste

Low-level radioactive waste would be generated as a result of decladding and cleaning blanket spent nuclear fuel and electrometallurgically treating driver spent nuclear fuel at ANL-W. This would result from activities in the Fuel Conditioning Facility and the Hot Fuel Examination Facility at ANL-W (e.g., equipment decontamination and repair), as well as in other facilities at ANL-W (e.g., analytical laboratory activities). Material in this waste stream has been generated and routinely handled at ANL-W for many years.

The volume of low-level radioactive waste resulting from decladding and cleaning of blanket spent nuclear fuel and electrometallurgical treating driver spent nuclear fuel at ANL-W that would require disposal (after volume reduction) would be approximately 50 cubic meters per year (1,766 cubic feet per year). This represents approximately 1 percent of the total annual volume of low-level radioactive waste currently being disposed of at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex, and the total of 555 cubic meters (19,600 cubic feet) represents approximately 0.7 percent of the total Radioactive Waste Management Complex disposal capacity.

Other Associated Process Transuranic Waste

Transuranic waste would be generated from decontamination activities for repair and maintenance of items, and miscellaneous work associated with processing the sodium-bonded spent nuclear fuel. Transuranic waste would be generated primarily from activities conducted in gloveboxes and hot cells at ANL-W.

All of the transuranic waste generated at ANL-W would be packaged and certified in accordance with the Waste Isolation Pilot Plant acceptance criteria prior to transport to the Waste Isolation Pilot Plant facility. The transuranic waste generated would amount to approximately 1 cubic meter per year (35 cubic feet per year), which is less than 0.002 percent of the volume of transuranic waste in retrievable storage at the Radioactive Waste Management Complex at INEEL. The total volume of transuranic waste would be approximately 9.1 cubic meters (321 cubic feet), which is approximately 0.005 percent of the estimated total volume of transuranic waste to be placed at the Waste Isolation Pilot Plant.

Other Associated Process Mixed Waste

Mixed waste would be generated primarily from the disposal of any cadmium-contaminated equipment or cleanup material and the analysis of cadmium samples. Mixed waste would be handled according to ANL-W

procedures that require limited accumulation at the point of generation. Interim storage of this waste would be accomplished at the Radioactive Scrap and Waste Facility prior to eventual disposal. The Radioactive Scrap and Waste Facility is a permitted mixed waste storage facility for these materials. The mixed waste streams that contribute to the overall mixed waste generated by electrometallurgical treatment have been identified in the INEEL Site Treatment Plan (DOE 1995b).

Deactivation Waste

A variety of waste would be generated as part of deactivation activities at ANL-W. This would include process equipment and process material such as electrorefiner salt and cadmium from electrometallurgical treatment of driver spent nuclear fuel. Generated waste categories would include low-level radioactive waste, transuranic waste, and mixed waste. This waste would be categorized and disposed of according to DOE Orders and ANL-W radioactive waste management procedures, as described above for each waste category.

The largest volume of deactivation waste would be low-level radioactive waste generated as a result of equipment dismantling and disposal. Components of electrometallurgical treatment of the driver spent nuclear fuel that would require disposition include two electrorefiners, two hot hydrostatic presses, and two V-mixers, as well as other components such as the grinder/crusher. Deactivation of these components would generate additional mixed, transuranic, and low-level radioactive waste that would require management.

The deactivation waste volume would be generated over a period of two years. The total deactivation waste would represent an additional 30 percent over the total associated process waste requiring disposal.

SRS

The melt and dilute process at SRS would generate process waste from treatment operations and other associated process waste from support operations. Process waste would include metallic high-level radioactive waste. Other associated process waste would include operational waste such as failed equipment, rags, packaging materials, and other miscellaneous items. The associated process waste would include low-level radioactive waste, transuranic waste, and mixed waste. All of the waste streams would be categorized according to existing DOE Orders and SRS waste management procedures. The anticipated categorization of the waste types and their expected interim storage and final disposal locations are given in **Table 4–55**.

Table 4–55 Waste Material Categories at SRS and Interim and Final Locations

Waste Stream	*Category*	*Interim Storage Location*	*Final Disposal Location*
Process Waste			
Melt and dilute product	High-level radioactive metallic waste	L-Area	Geologic repository
Off-gas filters	High-level radioactive waste [a]	L-Area	Geologic repository
Other Associated Process Waste			
Less than 100 nanocuries per gram transuranic waste Contaminated	Low-level radioactive waste	None	Low-activity waste vaults
Greater than 100 nanocuries per gram transuranic waste	Transuranic waste	Transuranic waste storage pads	Waste Isolation Pilot Plant
Offsite	Mixed waste	Mixed waste storage buildings	Offsite

[a] Cleaning of the contaminated filters would generate high-level radioactive liquid waste.

Estimates of the amounts of waste generated as a result of the melt and dilute processing at SRS are provided in **Table 4–56**. These values are based on an evaluation of waste forecasts that accounts only for the fraction of total waste that would be attributable to processing the blanket spent nuclear fuel pins.

Table 4–56 Amounts of Waste Generated at SRS Under Alternative 5

Waste Stream	*Waste Quantities (cubic meters)* [a]
Direct Process Waste	
Melt and dilute product	76 (189 canisters)[b]
Liquid high-level radioactive waste	30[c]
Saltstone [d]	78
Other Associated Process Waste	
Low-level radioactive waste	330[e]
Transuranic waste	16.5
Mixed waste	3

[a] Except for the number of canisters of melt and dilute products, the values given are estimated based on the heavy metal mass ratio of similar material processed at SRS (20 metric tons of heavy metal) and provided in DOE 2000.
[b] Standardized spent nuclear fuel canisters.
[c] This is a liquid high-level radioactive waste volume which results in about one Defense Waste Processing Facility borosilicate glass high-level radioactive waste canister or a solid high-level radioactive waste volume of 0.62 cubic meters.
[d] This is a secondary process waste from processing the high-level radioactive waste.
[e] Final disposal volume following a volume reduction (a reduction factor of 4 was assumed).

As indicated in the following waste-type discussions, the amounts of waste associated with this processing alternative are relatively small compared to onsite and offsite management capacities.

Direct Process Waste

During the melt and dilute process, a high-level radioactive waste melt and dilute product (metallic waste) would be the primary product. This waste would be temporarily stored in L-Area prior to ultimate disposition in an offsite (proposed geologic) repository. In addition, some high-level radioactive waste would be generated from cleaning the off-gas filter system, which contains cesium, tellurium, and other isotopes volatilized during the melt and dilute process. The high-level radioactive waste would be processed in the Defense Waste Processing Facility to yield vitrified high-level radioactive waste and saltstone. The vitrified high-level radioactive waste would be temporarily stored at the Defense Waste Processing Facility pending ultimate disposal in a geologic repository. The saltstone is a cement form of low-level radioactive waste that is generated as a by-product of SRS high-level radioactive waste tank form operations. The saltstone would be disposed of on site in the Z-Area Saltstone Vaults. The volume of this saltstone would be about 0.0070 percent of the 1.11 million-cubic meter (39.2 million-cubic foot) disposal capacity of the low-activity waste vaults.

Other Associated Process Low-Level Radioactive Waste

Low-level radioactive waste would be generated during the melt and dilute process. The volume of low-level radioactive waste resulting from this alternative (after volume reduction) would be about 1.1 percent of the total 30,500-cubic meter (1.08 million-cubic foot) disposal capacity of the low-activity waste vaults.

Other Associated Process Transuranic Waste

The volume of transuranic waste that would be generated during the melt and dilute process would be about 0.01 percent of the current 168,500-cubic meter (5.95 million-cubic foot) limit for the Waste Isolation Pilot Plant (DOE 1997).

Other Associated Process Mixed Waste

Mixed waste that would be generated during the melt and dilute process would be temporarily stored on site in the Mixed Waste Storage Buildings prior to eventual offsite disposal. The volume of this waste would be about 0.16 percent of the 1,900-cubic meter (67,100-cubic foot) storage capacity of these storage buildings.

4.8 ALTERNATIVE 6: MELT AND DILUTE BLANKET AND DRIVER FUEL AT ANL-W

Under this alternative, sodium-bonded blanket and driver spent nuclear fuel would be treated using the melt and dilute process at ANL-W. The melt and dilute product generated by this treatment process would be stored at the Radioactive Scrap and Waste Facility pending repackaging and transportation for disposal in a geologic repository. Both blanket and driver spent nuclear fuel would be cleaned to remove metallic sodium to the extent possible. The removed sodium would be stabilized using an oxidation/carbonation process (ANL 1999). The preparation of driver and blanket spent nuclear fuel to remove metallic sodium could start in 2003. The treatment of driver and blanket spent nuclear fuel by melt and dilute processing at ANL-W could start in 2005 and could be completed by 2015. In addition, a full year of operations would be needed to deactivate the processing equipment and establish an industrially safe configuration.

4.8.1 Air Quality

Nonradiological Gaseous Emissions

It is expected that criteria and hazardous air pollutants released from operational activities at ANL-W under this alternative would be the same as for Alternative 1, with the exception that there would be no cadmium release, as described in Section 4.3.1 (see also Appendix E, Section E.5.3.1, for more detail). Baseline air quality concentrations are presented in Section 3.2.3.1.

Radiological Gaseous Emissions

The cleaning of the blanket and driver spent nuclear fuel and the melt and dilute treatment of this fuel would release gaseous fission products to the hot argon cell environment. Krypton-85 and elemental tritium would be the most prevalent radioactive gaseous fission products released to the environment. The tritium released into the cell would not be oxidized due to a very low presence of oxygen and humidity in the argon cell. The argon cell also would contain an equilibrium concentration of other radionuclide isotopes. Appendix E, Section E.4.1, provides a list of various isotopes that would be present in the argon cell in nanocuries (10^{-9} curies) and would be released to the atmosphere through the facility stack, along with krypton-85 and elemental tritium. The maximum release of radioactive gases would occur when cutting of blanket and driver spent nuclear fuel to remove metallic sodium is performed simultaneously. This simultaneous operation could occur over a 2-year period during the estimated 10 years of operation, starting in 2003. Appendix E, Section E.4.5, provides more details on various releases during the processing period at ANL-W. Based on an annual processing throughput of 10 metric tons of heavy metal of blanket spent nuclear fuel elements and about 1.7 metric tons of heavy metal of driver spent nuclear fuel elements, about 2,162 curies of elemental tritium and 32,650 curies of gaseous krypton-85 would be released annually to the atmosphere (see Appendix E, Section E.4.5).

4.8.2 Water Resources

Surface Water

No surface water is used at ANL-W. Flood waters from the Big Lost River would not be expected to reach the facilities at ANL-W as shown in Figure 3–3.

Nonradiological Liquid Effluent

There are no discharges to the surface waters at ANL-W, except for discharges of nonhazardous liquid waste to the sewage pond and the industrial waste pond. Big Lost River, Little Lost River, and Birch Creek would not be impacted by activities associated with the melt and dilute treatment process. Current operating and monitoring practices would continue for stormwater and liquid effluent discharges associated with facilities at ANL-W (see also Section 4.2.2).

During fuel treatment and associated activities, some hazardous materials may be used inside buildings. To prevent potential releases to surface or subsurface waters resulting from spills of hazardous materials used in buildings, these facilities are designed, constructed, and maintained to contain these materials. Double-contained pipes, leak detection, and secondary containment of tanks are some of the features used to prevent hazardous materials from being released to the environment. Following existing written procedures, spill containment and cleanup equipment is present in areas where hazardous materials are stored or used (DOE 1996b).

Radiological Liquid Effluent

No radiological liquid effluent generated by melt and dilute treatment process operations would be discharged to surface water at ANL-W.

Groundwater

Under this alternative at ANL-W, there would be little change in groundwater consumption for domestic use since there is little change expected in the number of workers. Water consumption for the melt and dilute treatment process operations would not impact the current water usage at ANL-W. The current water usage at ANL-W is 188 million liters per year (49.6 million gallons per year).

Nonradiological Liquid Effluent

No nonradiological liquid effluent generated by melt and dilute treatment process operations would be discharged to groundwater.

Radiological Liquid Effluent

No radiological liquid effluent generated by melt and dilute treatment process operations would be discharged to groundwater.

4.8.3 Socioeconomics

Under this alternative, the existing facilities at ANL-W would remain operational. No new employment or in-migration of workers would be required. Thus, there would be no additional impacts on the socioeconomic conditions in the region around INEEL.

4.8.4 Public and Occupational Health and Safety

The assessments of potential incremental radiological and chemical impacts associated with this alternative are presented in this section. Summaries of radiological impacts from normal operations are presented in Tables 4–57 and 4–58 for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in Tables 4–59 and 4–60. The impacts from hazardous chemical

releases during accident conditions are presented in Table 4–61. Background information on the effects of radiation on human health and safety is presented in Appendix E, Section E.2.

4.8.4.1 Normal Operations

Radiological Impacts

Under this alternative, radioactive releases would occur during sodium-bonded blanket spent nuclear fuel and driver spent nuclear fuel cleaning and melt and dilute processes. All of these activities would be performed in the argon cell. Appendix E, Sections E.3, E.4.1, and E.4.2, details the method and assumptions used for calculating the impacts of normal operational radiological releases on the public health and safety. The maximum annual dose to the public would result when both blanket and driver spent nuclear fuel are treated simultaneously under this alternative. Appendix E, Section E.4.5, provides the details on treatment process duration and throughputs for each fuel type.

Calculated maximum annual and project total radiological impacts to the public are given in **Table 4–57**. The impacts are calculated for two types of receptors: the general public living within 80 kilometers (50 miles) of ANL-W in the year 2010, and a maximally exposed offsite individual (a member of the public assumed to be residing at the INEEL site boundary and receiving the maximum dose). Primary contributors to doses to members of the public would be from releases of tritium gases (about 1 percent of which were assumed conservatively to be in oxidized form) and krypton-85; together they contribute over 99.9 percent of the total calculated doses. To put the operational impacts into perspective, comparisons with impacts from natural background radiation also are included in the table. As shown in this table, the expected radiation doses to the maximally exposed offsite individual and the general public are much smaller than the limit of 10 millirem per year set by the EPA (40 CFR 61) and DOE (DOE Order 5400.5).

Table 4–58 summarizes worker population doses. Occupational doses were estimated by examining the type and duration of various operations performed by the ANL-W workers involved with the melt and dilute treatment processes. It was concluded that the average worker dose would not be different from that currently being experienced. The estimated annual collective worker dose would be 22 person-rem, with an average individual dose of 60 millirem per year for each of the 346 involved workers. If these estimates were extended over the 12 years of treatment activities (assuming operations would start in 2003 and end in 2015) and the 33 person-rem from 1 year of deactivation activities were included, the project total worker population dose would be 297 person-rem, leading to a risk of 0.12 latent cancer fatalities.

As shown in Tables 4–57 and 4–58:

- The annual dose to the maximally exposed offsite individual would be 0.0020 millirem per year, with an associated risk of developing a fatal cancer of 1×10^{-9} per year (or one chance in 1 billion that the individual would develop a fatal cancer per year of exposure).

- The collective dose to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 0.012 person-rem per year, with an associated 6×10^{-6} latent cancer fatalities per year (or one chance in 167,000 that the population would develop a fatal cancer per year of exposure).

- The project total risk to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 0.000012 latent cancer fatalities (or one chance in 83,000 that the exposed population would develop a fatal cancer).

Table 4–57 Annual and Project Total Radiological Impacts to the Public From Operational Activities Under Alternative 6

Receptor	*Melt and Dilute Driver Spent Nuclear Fuel at ANL-W*	*Melt and Dilute Blanket Spent Nuclear Fuel at ANL-W*	*Total*
Population Within 80 Kilometers (50 Miles) in the Year 2010			
Collective dose (person-rem) [a]	0.012	0.00028	0.012
Excess latent cancer fatalities (per year)	6.0×10^{-6}	1.4×10^{-7}	6.1×10^{-6}
Project total excess latent cancer fatalities [b]	0.000012	3.4×10^{-7}	0.000012
Maximally Exposed Offsite Individual			
Dose (millirem per year) [a]	0.002	0.000048	0.0020
Percent of annual background radiation [c]	0.00056	0.000013	0.00057
Latent cancer fatality risk (per year)	1.0×10^{-9}	2.4×10^{-11}	1.0×10^{-9}
Project total lifetime cancer fatality risk	2.0×10^{-9}	5.8×10^{-11}	2.0×10^{-9}
Average Individual Within 80 Kilometers (50 Miles)			
Dose (millirem per year) [d]	0.00005	1.2×10^{-6}	0.000051
Latent cancer fatality risk (per year)	2.5×10^{-11}	5.8×10^{-13}	2.6×10^{-11}
Project total lifetime cancer fatality risk [b]	5.0×10^{-11}	1.4×10^{-12}	5.1×10^{-11}

[a] Annual maximum dose during normal operations.
[b] Total calculated dose over 12 years. Nearly all of the impacts are associated with releases of tritium, krypton, and iodine that would occur during the cleaning process. The impact of releases resulting from melt and dilute processing only are not significant.
[c] The annual natural background radiation level at INEEL is about 360 millirem for the average individual (see Table 3–8); the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 person-rem.
[d] Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of ANL-W in the year 2010 (240,338).

Table 4–58 Annual and Project Total Radiological Impacts to Workers From Operational Activities Under Alternative 6

Receptor	*Impacts*
Worker [a]	
Average worker dose (millirem per year)	60
Average worker latent cancer fatality risk (project total over 12 years)	0.00029
Worker Population	
Collective dose (person-rem per year) [b]	22
Excess latent cancer fatalities (per year)	0.0088
Project total dose (person-rem)	297
Project total excess latent cancer fatalities	0.12

[a] The regulatory dose limit for an individual worker is 5,000 millirem per year (10 CFR 835). However, the maximum annual dose to a worker would be kept below the DOE Administrative Control Level of 2,000 millirem per year, as established for all DOE activities in DOE Order N 441.1.
[b] Increases to 33 person-rem for one year of deactivation activities.

Source: ANL 1999.

- The collective dose to facility workers would be 22 person-rem per year, with an associated 0.0088 latent cancer fatalities (or one chance in 113 that the workers would experience a fatal cancer per year of operation).

- The project total dose to facility workers would be 297 person-rem with an associated 0.12 latent cancer fatalities (or one chance in eight that the exposed workers would experience a fatal cancer).

These results indicate that no increase in the expected number of fatal cancers among the various affected populations would result from the activities under this alternative.

Hazardous Chemical Impacts

The hazardous chemical impacts to the public residing at the INEEL site boundary from the operational activities at ANL-W under Alternative 6 would be similar to the impacts evaluated for Alternative 1, with the exception that there would be no cadmium release, as described in Section 4.3.4.1. The results indicate that no adverse toxic health or cancer effects would be expected from exposure to hazardous chemicals released under this alternative. The existing chemical environment is presented in Section 3.2.10.2.

4.8.4.2 Facility Accidents

Radiological Impacts

The potential radiological impacts to the public and a noninvolved onsite worker due to accidents during operational activities related to melt and dilute processing of fuel elements are summarized and presented in this section. The detailed analysis of facility accidents and the associated assumptions are presented in Appendix F. The detailed analysis considered a wide spectrum of potential accident scenarios, including fire, spills, criticality, earthquake, and aircraft crash. The aircraft crash event was determined to have an occurrence frequency of less than 10^{-7} per year, and consequence analyses for this event were not performed. Double-batching of the driver spent nuclear fuel was determined potentially to result in a criticality event (see Appendix F), and this event was analyzed for the driver spent nuclear fuel only. Processing of the blanket and driver spent nuclear fuel would be performed in the Hot Fuel Examination Facility. The multifacility impacts of the beyond-design-basis earthquake are not relevant to processing of the fuel under this option. Therefore, only the higher frequency design-basis earthquake event was analyzed. **Table 4–59** presents the frequencies and consequences of the postulated set of accidents to the maximally exposed offsite individual; the offsite population residing within 80 kilometers (50 miles) of the facility; and a noninvolved worker located 100 meters (330 feet) to 230 meters (755 feet) from the facility. The 230-meter (755-foot) distance is the ANL-W bus staging area, which leads to a higher dose to the noninvolved worker for the scenarios with elevated releases.

The dose to the maximally exposed offsite individual was calculated for the 95th percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50th percentile meteorological conditions. DOE did not quantitatively estimate the involved worker dose due to accidents. (See the discussion on the involved worker in Section 4.2.4.2.) The accident risks for the same receptors are summarized in **Table 4–60**.

For accidents at ANL-W, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and a noninvolved worker would be 0.000076 per year (or one chance in 13,160 that the individual would develop a fatal cancer per year of operation) and 2.7×10^{-6} per year (or one chance in 370,400 that the individual would develop a fatal cancer per year of operation), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.0090 per year (or one chance in 111 that the population would experience a fatal cancer per year of operation).

Table 4–59 Accident Frequency and Consequences Under Alternative 6

Accident	Frequency (event per year)	Maximally Exposed Offsite Individual		Population Within 80 Kilometers (50 Miles)		Noninvolved Worker	
		Dose (millirem)	Latent Cancer Fatality Risk [a]	Dose (person-rem)	Excess Latent Cancer Fatalities [b]	Dose (millirem)	Latent Cancer Fatality Risk [a]
Driver Spent Nuclear Fuel							
Waste handling spill	0.0024	597	0.00030	70.8	0.035	26.7	0.000011
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Cask drop	0.01	0.030	1.5×10^{-8}	0.0035	1.8×10^{-6}	0.00084	3.4×10^{-10}
Design-basis earthquake	0.008	19000	0.0095	2250	1.1	840	0.00034
Sodium fire[c]	0.008	282	0.00014	33	0.016	2.6	1.0×10^{-6}
Criticality	0.0003	0.52	2.6×10^{-7}	0.085	0.000043	0.47	1.9×10^{-7}
Blanket Spent Nuclear Fuel							
Waste handling spill	0.0024	15	7.5×10^{-6}	1.8	0.00090	0.49	2.0×10^{-7}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Cask drop	0.01	0.0024	1.2×10^{-9}	0.00028	1.4×10^{-7}	0.000049	2.0×10^{-11}
Design-basis earthquake	0.008	471	0.00024	56.1	0.028	15.2	6.1×10^{-6}
Sodium fire [c]	0.008	5.9	3.0×10^{-6}	0.69	0.00035	0.054	2.2×10^{-8}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.
[c] The frequency for this event is the frequency for the facility design-basis earthquake-initiating cell fire.

Table 4–60 Annual Cancer Risks Due to Accidents at ANL-W Under Alternative 6

Accident	Maximally Exposed Offsite Individual [a]	Population Within 80 Kilometers (50 Miles) [b]	Noninvolved Worker [a]
Driver Spent Nuclear Fuel			
Waste liquid spill	7.2×10^{-7}	0.000085	2.6×10^{-8}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Cask drop	1.5×10^{-10}	1.7×10^{-8}	3.4×10^{-12}
Design-basis earthquake	0.000076	0.0090	2.7×10^{-6}
Sodium fire	1.1×10^{-6}	0.00013	8.3×10^{-9}
Criticality	8.0×10^{-11}	1.3×10^{-8}	5.7×10^{-11}
Blanket Spent Nuclear Fuel			
Waste liquid spill	1.8×10^{-8}	2.2×10^{-6}	4.8×10^{-10}
Cask drop	1.2×10^{-11}	1.4×10^{-9}	2.0×10^{-13}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Design-basis earthquake	1.9×10^{-6}	0.00023	4.8×10^{-8}
Sodium fire	2.4×10^{-8}	2.8×10^{-6}	1.7×10^{-10}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

Hazardous Chemical Impacts

Nonradiological impacts are evaluated in terms of comparison to ERPG. ERPG values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects (see Appendixes F, Section F.3.1.2 for details).

The hazardous chemical impacts of potential facility accidents associated with the treatment of the driver spent nuclear fuel using the electrometallurgical process are summarized in **Table 4–61**.

Table 4–61 Hazardous Chemical Impacts Due to Accidents at ANL-W Under Alternative 6

Accident	*Frequency (event per year)*	*Receptor*	*Exposure*
Sodium fire	0.008	Noninvolved worker	Sodium: less than ERPG-1
		Maximally exposed offsite individual	Sodium: less than ERPG-1

ERPG = Emergency Response Planning Guideline.

4.8.5 Environmental Justice

As discussed in Appendix H, Executive Order 12898 directs Federal agencies to address disproportionately high and adverse health or environmental effects of alternatives on minority or low-income populations.

Analyses of normal operations and accident conditions presented in Section 4.8.4 show the risk of latent cancer fatalities to the public residing within 80 kilometers (50 miles) of the melt and dilute treatment processing facilities at ANL-W to be much lower than 1. Therefore, radiological and nonradiological risks posed by implementation of this alternative would have no disproportionately high and adverse consequences on any particular group within the general population, including minority or low-income populations.

4.8.6 Waste Management

This alternative would generate process waste from treatment operations, other associated process waste from normal support operations, and deactivation waste following the conclusion of operations. Process waste would include high-level radioactive metallic and ceramic waste from stabilizing the residual waste from the existing Electrometallurgical Demonstration Project. Other associated process waste would include operational waste such as failed equipment, rags, packaging materials, and other miscellaneous items. Deactivation waste would include the disposal of process equipment and other materials. All of these materials would be categorized according to existing DOE Orders and ANL-W waste management procedures. The anticipated categorization of these waste types and their expected interim storage and final disposal locations are given in Table 4–9 (see Section 4.2.6). The quantities of various waste forms generated as a result of Alternative 6 are provided in **Table 4–62**.

Estimates of the total amount of other associated process waste generated are based on an evaluation of waste forecasts from ANL-W, together with an understanding of melt and dilute process activities resulting in the generation of each waste category. The values in Table 4–62 are for disposal and include volume reduction. It is anticipated that a large fraction of the low-level radioactive waste that would be generated as a result of Alternative 6 could be volume-reduced by up to 100 percent at the Waste Experimental Reduction Facility at INEEL prior to disposal at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex.

Table 4–62 Amounts of Waste Generated at ANL-W Under Alternative 6[a]

Waste Stream	Waste Quantities	
	Volume (cubic meters)	Mass (kilograms)
Direct Process Waste		
Fuel assembly hardware (low-level radioactive waste)	0	0
High-level radioactive ceramic waste	19.4 (31 canisters) [b]	29,000
High-level radioactive metallic waste	0.6 (1 canister) [b]	460
Melt and dilute product	65.6 (164 canisters) [b]	136,400
Other Associated Process Waste		
High-level radioactive waste	0.4 (1 canister) [b]	220
Low-level radioactive waste [c]	711	144,000
Transuranic waste	12.5	5,400
Mixed waste	35.3	19,000
Sanitary waste	4,960	1.72×10^6
Deactivation Waste		
Low-level radioactive waste [c]	213	72,000
Transuranic waste	1.6	853
Mixed waste	5.9	3,500

[a] These waste generation estimates are through the year 2015. This is the assumed date that these materials might be sent to the repository. Treatment, high-level radioactive waste processing, deactivation, and interim storage would be accomplished during this time period.

[b] Standardized canisters.

[c] The volumes listed represent final disposal volumes following volume reduction at the Waste Experimental Reduction Facility at INEEL.

Source: ANL 1999.

The waste values in Table 4–62 represent total quantities that would be produced as a result of Alternative 6 operations. They are not incremental increases over the volumes provided in Table 4–10 that would result from the No Action Alternative. In Alternative 6, both the driver and blanket spent nuclear fuel would be transformed into a high-level radioactive waste form (melt and dilute product) for disposal in the repository, and in this conversion process, the total volume of material to be disposed of in the repository would be reduced from direct disposal values.

Direct Process Waste

For this alternative, fuel hardware would be used as part of the stainless steel to form the metal ingot for disposal of the fuel by melting. Its mass is included as part of the spent nuclear fuel disposal.

Under this alternative, metallic and ceramic high-level radioactive waste would be produced from existing process material at ANL-W. This waste would be generated to stabilize materials produced during the electrometallurgical demonstration project. In addition, the salt removed from the melting furnace used for driver spent nuclear fuel would contain fission products that would be stabilized in ceramic waste. The volumes of waste forms provided in Table 4–62 are for the standardized canisters required for disposal of these materials.

A second metallic high-level radioactive waste called the melt and dilute product would be generated as a result of the melt and dilute treatment of fuel at ANL-W. This waste, along with the ceramic and metallic waste from the demonstration project, would be stored temporarily at the Radioactive Scrap and Waste Facility at ANL-W to allow retrieval for future disposal. The Radioactive Scrap and Waste Facility was designed and

constructed for temporary storage of this type of waste. Shielding would be provided by a combination of (1) steel storage liners which would store the waste, and (2) the soil surrounding the liners. When a geologic repository is available, the waste cans containing this high-level radioactive waste would be removed from storage, shipped to the INEEL Dry Transfer Facility, and prepared for shipment to the repository.

Other Associated Process High-Level Radioactive Waste

High-level radioactive waste would be generated as a result of driver and blanket spent nuclear fuel processing at ANL-W. This would result from activities in the Hot Fuel Examination Facility. Material in this waste stream would consist of the absorbent used in the off-gas system which would have collected the volatile radionuclides released from the spent nuclear fuel when heated.

The volume of high-level radioactive waste generated is expected to be less than the amount needed to fill a single standardized waste canister. Conservatively, the volume of a single canister, 0.4 cubic meters, was used for the volume of high-level radioactive waste generated.

Other Associated Process Low-Level Radioactive Waste

Low-level radioactive waste would be generated as a result of the melt and dilute treatment of fuel at ANL-W. This would result from activities in the Fuel Conditioning Facility and the Hot Fuel Examination Facility at ANL-W (e.g., equipment decontamination and repair), as well as in other facilities at ANL-W (e.g., analytical laboratory activities). Material in this waste stream has been generated and routinely handled at ANL-W for many years.

The volume of low-level radioactive waste at ANL-W that would require disposal (after volume reduction) would be approximately 50 cubic meters (1,766 cubic feet) per year. This represents approximately 1 percent of the total annual volume of low-level radioactive waste currently being disposed of at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex, and the total of 711 cubic meters (25,100 cubic feet) represents approximately 0.6 percent of the total Radioactive Waste Management Complex disposal capacity.

Other Associated Process Transuranic Waste

Transuranic waste would be generated from decontamination activities for repair and maintenance of items, and miscellaneous work associated with processing the sodium-bonded spent nuclear fuel. Transuranic waste would be generated primarily from activities conducted in gloveboxes and hot cells at ANL-W.

All of the transuranic waste generated at ANL-W would be packaged and certified in accordance with Waste Isolation Pilot Plant acceptance criteria prior to transport to the Waste Isolation Pilot Plant. The transuranic waste generated would amount to approximately 1 cubic meter (35 cubic feet) per year, which is less than 0.002 percent of the volume of transuranic waste in retrievable storage at the Radioactive Waste Management Complex at INEEL. The total volume of transuranic waste would be 12.5 cubic meters (441 cubic feet), which is less than 0.008 percent of the estimated total volume of transuranic waste to be placed at the Waste Isolation Pilot Plant.

Other Associated Process Mixed Waste

Mixed waste would be generated primarily from the disposal of any cadmium-contaminated equipment or cleanup material and the analysis of cadmium samples. Mixed waste would be handled according to ANL-W procedures that require limited accumulation at the point of generation. Interim storage of this waste would be accomplished at the Radioactive Scrap and Waste Facility prior to eventual disposal. The Radioactive

Scrap and Waste Facility is a permitted mixed waste storage facility for these materials. The mixed waste streams that contribute to the overall mixed waste generated by electrometallurgical treatment are identified in the INEEL Site Treatment Plan (DOE 1995b).

Deactivation Waste

A variety of waste would be generated as part of deactivation activities associated with melt and dilute treatment of sodium-bonded spent nuclear fuel at ANL-W. Generated waste categories would include low-level radioactive waste, transuranic waste, and mixed waste. This waste would be categorized and disposed of according to DOE Orders and ANL-W radioactive waste management procedures, as described above for each waste category.

The largest volume of deactivation waste would be low-level radioactive waste generated as a result of dismantling and disposing of electrometallurgical treatment and melt and dilute processing equipment. Components of the electrometallurgical demonstration project that would require disposition include two electrorefiners; two hot hydrostatic presses; and one V-mixer, as well as other components such as the grinder/crusher. Deactivation of components would generate additional mixed, transuranic, and low-level radioactive waste that would require management. The total deactivation waste would represent 35 percent over the total associated process waste requiring disposal.

4.9 Transportation Impacts

Transportation impacts may be divided into two parts: the impacts of incident-free or routine transportation, and the impacts of transportation accidents. Incident-free transportation and transportation accident impacts are divided into two parts: nonradiological impacts and radiological impacts. Incident-free transportation impacts include radiological impacts on the public and the crew from the radiation field that surrounds the package. Nonradiological impacts of incident-free transportation are from vehicular emissions. Nonradiological impacts of potential transportation accidents include traffic accident fatalities. Only in the worst conceivable conditions, which are of low probability, could a transportation cask of the type used to transport radioactive material be so damaged that a release of radioactivity to the environment could occur.

The impact of a specific accident is expressed in terms of probabilistic risk, which is the probability of that accident occurring multiplied by its consequence. Hypothetical accidents ranging from a low-speed impact to those involving high-speed impacts with or without fires leading to cask failure are analyzed. The accident frequencies and consequences are evaluated using the method developed for the NRC, which is known as the "Modal Study" (NRC 1987). The overall risk is obtained by summing the individual risks from all accident conditions. The risks for radiological accidents are expressed as additional latent cancer fatalities and as additional immediate fatalities for nonradiological accidents. The risks of incident-free effects are expressed in additional latent cancer fatalities.

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 materials. Calculation of risk factors was accomplished by using the HIGHWAY (Johnson et al. 1993) computer codes to choose representative routes according to U.S. Department of Transportation regulations. These codes provide population estimates so that RADTRAN 5 (Neuhauser and Kanipe 1998) codes could be used to determine the radiological risk factors. This analysis is discussed in Appendix G. **Table 4–63** lists the sodium-bonded spent nuclear fuel destinations evaluated in this EIS.

Table 4–63 Transportation Summary for Sodium-Bonded Fuel

Fuel Type	*Alternatives* [a]	*Metric Tons of Heavy Metal*	*Origin/State*	*Destination/ State*	*Cask*	*Number of Shipments/Type of Transport*
EBR-II driver	All	1.1	ANL-W/ID	ANL-W/ID	HFEF-5	84/On site, intrafacility transfers
EBR-II driver	All	2.0	INTEC/ID	ANL-W/ID	TN-FSV or NAC-LWT	17/On site with roads open or 43/On site with roads open
EBR-II blanket	All	22.4	ANL-W/ID	ANL-W/ID	HFEF-5	165/On site, intrafacility transfers
Fast Flux Test Facility driver [b]	All	0.33	Hanford/WA	ANL-W/ID	T-3	10/Public highways
Fermi-1 blanket	All	34.2	INTEC/ID	ANL-W/ID	PB-1	14/On site with road closed
Miscellaneous [b]	All	0.1	Oak Ridge National Laboratory/TN, Sandia National Laboratories/NM, SRS/SC	ANL-W/ID	To be determined by DOE	1/Public highways 1/Public highways 1/Public highways
Declad EBR-II blanket	3 and 5	22.4	ANL-W/ID	SRS/SC	NAC-LWT	11/Public highways
Declad Fermi-1 blanket	3 and 5	34.2	ANL-W/ID	SRS/SC	NAC-LWT	18/Public highways

ID = Idaho; NM = New Mexico; SC = South Carolina; TN = Tennessee; WA = Washington

[a] "All" includes the proposed action plus the No Action Alternative.

[b] This fuel is assumed to be in Idaho per the amended Record of Decision for the Programmatic Spent Nuclear Fuel EIS (61 FR 9441).

Transportation of the Fast Flux Test Facility driver spent nuclear fuel currently stored at the Hanford site and the small amounts of miscellaneous sodium-bonded spent nuclear fuel currently stored at Oak Ridge National Laboratory, Sandia National Laboratories, and at SRS (see Section 2.2.3 and Appendix D, Section D.5.2, for more details on miscellaneous fuel types) are shipment campaigns related to sodium-bonded spent nuclear fuel and were analyzed by DOE in the Programmatic Spent Nuclear Fuel EIS (DOE 1995a and 61 FR 9441), so they are included by reference in this impact analysis. See Appendix G for more details.

All EBR-II blanket and some EBR-II driver spent nuclear fuel are currently stored at ANL-W and would be subject to a building-to-building movement for processing. Since the movement is a short distance on closed DOE-controlled roads, DOE procedures and the NRC regulations do not require the use of a certified Type B cask. No incident-free risk analysis is necessary because the public would receive no measurable exposure. The worker dose is included in the process and handling dose estimates because the same personnel would be moving the spent nuclear fuel. The probability and consequences of potential accidents during movement are bounded in frequency and consequence by handling accidents.

Fermi-1 blanket spent nuclear fuel would be shipped from INTEC to ANL-W in the Type B cask (PB-1 Cask). Since DOE would close the roads between INTEC and ANL-W using existing traffic gates, and the road is uninhabited, no quantitative analysis is necessary. No incident-free risk analysis is necessary because the public would receive no measurable exposure. The worker dose is included in the process and handling dose estimates because the same personnel would be moving the spent nuclear fuel. Once the cask is closed for

movement on the closed roads, the likelihood and consequences of any foreseeable accident would be very small and are not further quantified.

EBR-II driver spent nuclear fuel would be shipped from INTEC to ANL-W in a certified Type B cask, either model TN-FSV or model NAC-LWT. Since the cask would be certified, DOE would not close the roads between INTEC and ANL-W. However, since the road is uninhabited, limited quantitative analysis is necessary. No incident-free risk analysis for exposure to the public at stops or in their homes is necessary. The worker dose is analyzed for the transportation crew, and the dose to other vehicles using the road is estimated. No accident analysis is necessary because potential accidents during movement are bounded in frequency and consequence by the handling accidents. Once the cask is closed for movement on the closed roads, the likelihood and consequences of any foreseeable accident would be very small.

Transportation-related risks are calculated and presented separately for workers and members of the general public. The workers considered are truck crew members involved in the actual overland transportation. The general public includes all persons who could be exposed to a shipment while it is moving or stopped during transit. The affected population includes individuals living within 800 meters (0.5 miles) of each side of the road. Potential risks are estimated for the affected population and for the hypothetical maximally exposed offsite individual. For incident-free operations, the maximally exposed individual would be an individual stuck in traffic next to the shipment for 30 minutes. For accident conditions, the maximally exposed individual is assumed to be an individual located 33 meters (105 feet) directly downwind from the accident. The risk to the affected population is a measure of the radiological risk posed to society as a whole by the alternative being considered. The impact to the affected population is used as the primary means of comparing various alternatives.

The following provides a summary of transportation impacts. Appendix G details the methods and assumptions used.

4.9.1 Onsite Transportation Impacts - No Action Alternative

Under all alternatives, EBR-II driver spent nuclear fuel would be shipped by DOE in 17 shipments using the TN-FSU cask, or 43 shipments using the NAC-LWT cask. The analysis assumes that 43 shipments are made. Fifteen ceramic waste form, 1 metallic waste form, and 355 spent nuclear fuel shipments would be made from ANL-W to the INEEL Dry Transfer Facility. The total distance traveled on public roads on the INEEL site by trucks carrying radioactive materials would be 16,000 kilometers (9,900 miles).

Impacts of Onsite Incident-Free Transportation. The dose to transportation workers from all transportation activities required by this alternative was estimated at 0.003 person-rem; the collective dose to the affected population would be 0.022 person-rem. Accordingly, incident-free transportation of radioactive material would result in 1.2×10^{-6} latent fatal cancers among transportation workers and 0.000011 latent fatal cancers in the total affected population over the duration of the transportation activities. Latent fatal cancers associated with radiological releases were estimated by multiplying the worker dose by 0.0004 latent fatal cancers per person-rem of exposure, and multiplying the collective dose to the affected population by 0.0005 latent fatal cancers per person-rem of exposure (ICRP 1991).

Impacts of Onsite Accidents During Ground Transportation. The maximum foreseeable onsite transportation accident under this alternative (probability of occurrence: more than 1×10^{-7} per year) would not breech the transportation cask. The probability of more severe accidents also was evaluated, and was estimated to be less than 1×10^{-7} per year.

Estimates of the total ground transportation accident risks under this alternative are as follows: a collective dose to the affected population of less than 1×10^{-6} person-rem would result in less than 1×10^{-9} latent fatal cancers; and traffic accidents would result in 0.00012 traffic fatalities.

4.9.2 Onsite Transportation Impacts - Alternative 1

In addition to the 43 shipments of EBR-II driver spent nuclear fuel, 125 ceramic waste form and 5 metallic waste form shipments from ANL-W to the INEEL Dry Transfer Facility would be made. The total distance traveled on public roads on the INEEL site by trucks carrying radioactive materials would be 6,700 kilometers (4,200 miles).

Impacts of Onsite Incident-Free Transportation. The dose to transportation workers from all transportation activities entailed by this alternative was estimated at 0.0044 person-rem; the collective dose to the affected population would be 0.033 person-rem. Accordingly, incident-free transportation of radioactive material would result in 1.8×10^{-6} latent fatal cancers among transportation workers and 0.000016 latent fatal cancers in the total affected population over the duration of the transportation activities.

Impacts of Onsite Accidents During Ground Transportation. The maximum foreseeable onsite transportation accident under this alternative (probability of occurrence: more than 1×10^{-7} per year) would not breech the transportation cask. The probability of more severe accidents also was evaluated, and was estimated to be less than 1×10^{-7} per year.

Estimates of the total ground transportation accident risks under this alternative are as follows: a collective dose to the affected population of less than 1×10^{-6} person-rem would result in less than 1×10^{-9} latent fatal cancers; and traffic accidents would result in 0.000052 traffic fatalities.

4.9.3 Onsite Transportation Impacts - Alternative 2

In addition to the 43 shipments of EBR-II driver spent nuclear fuel, 27 ceramic waste form, 7 metallic waste form, and 63 spent nuclear fuel shipments from ANL-W to the INEEL Dry Transfer Facility would be made. The total distance traveled on public roads on the INEEL site by trucks carrying radioactive materials would be 5,200 kilometers (3,200 miles).

Impacts of Onsite Incident-Free Transportation. The dose to transportation workers from all transportation activities entailed by this alternative was estimated at 0.0043 person-rem; the collective dose to the affected population would be 0.032 person-rem. Accordingly, incident-free transportation of radioactive material would result in 1.7×10^{-6} latent fatal cancers among transportation workers and 0.000016 latent fatal cancers in the total affected population over the duration of the transportation activities.

Impacts of Onsite Accidents During Ground Transportation. The maximum foreseeable onsite transportation accident under this alternative (probability of occurrence: more than 1×10^{-7} per year) would not breech the transportation cask. The probability of more severe accidents also was evaluated, and was estimated to be less than 1×10^{-7} per year.

Estimates of the total ground transportation accident risks under this alternative are as follows: a collective dose to the affected population of less than 1×10^{-6} person-rem would result in less than 1×10^{-9} latent fatal cancers; and traffic accidents would result in 0.00008 traffic fatalities.

4.9.4 On- and Offsite Transportation Impacts - Alternative 3

In addition to the 43 shipments of EBR-II driver spent nuclear fuel, Alternative 3 would require 11 shipments of declad EBR-II blanket material and 18 shipments of Fermi-1 blanket material from ANL-W to SRS. Twenty-seven ceramic waste form and 2 metallic waste form shipments from ANL-W to the INEEL Dry Transfer Facility would be made. The impacts for these alternatives include both on- and offsite transportation. The total distance traveled on public roads by trucks carrying radioactive materials would be 111,800 kilometers (69,500 miles).

Impacts of On- and Offsite Incident-Free Transportation. The dose to transportation workers from all transportation activities required by this alternative was estimated at 0.0052 person-rem; the collective dose to the affected population would be 0.042 person-rem. Accordingly, incident-free transportation of radioactive material would result in 2.1×10^{-6} latent fatal cancers among transportation workers and 0.000021 latent fatal cancers in the total affected population over the duration of the transportation activities. The dose to transportation workers from transporting cleaned and declad blanket spent nuclear fuel to SRS was estimated at 0.0012 person-rem; the collective dose to the affected population would be 0.012 person-rem. Accordingly, incident-free transportation of radioactive material would result in 4.7×10^{-7} latent fatal cancers among transportation workers and 6×10^{-6} latent fatal cancers in the total affected population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this alternative is 0.00039.

Impacts of On- and Offsite Accidents During Ground Transportation. The maximum foreseeable offsite transportation accident under this alternative (probability of occurrence: more than 1×10^{-7} per year) would be shipment of EBR-II blanket material from DOE's facility at ANL-W to SRS with a Severity Category 5 accident in a suburban population zone under neutral (average) weather conditions. The accident could result in a dose of 0.00024 person-rem to the public with an associated 1.2×10^{-7} latent fatal cancers, and 2.5×10^{-13} rem to the hypothetical maximally exposed individual with a latent fatal cancer risk of 1.3×10^{-15}. No fatalities would be expected. The probabilities of more severe accidents, different weather conditions at the time of accident, or occurrence in a more densely populated area also were evaluated, and were estimated to be less than 1×10^{-7} per year.

Estimates of the total ground transportation accident risks under this alternative are as follows: a collective dose to the affected population of 3.4×10^{-6} person-rem would result in 1.7×10^{-9} latent fatal cancers; and traffic accidents would result in 0.0018 traffic fatalities. Ground transportation accident risks to the affected population from transporting blanket fuel to SRS were estimated at 3×10^{-6} person-rem, resulting in 1.5×10^{-9} latent fatal cancers; and traffic accidents would result in 0.0017 traffic fatalities.

4.9.5 Onsite Transportation Impacts - Alternative 4

In addition to the 43 shipments of EBR-II driver spent nuclear fuel, 27 ceramic waste form, 2 metallic waste form, and 114 melt and dilute product shipments would be made from ANL-W to the INEEL Dry Transfer Facility. The total distance traveled on public roads on the INEEL site by trucks carrying radioactive materials would be 7,200 kilometers (4,500 miles).

Impacts of Onsite Incident-Free Transportation. The dose to transportation workers from all transportation activities required by this alternative was estimated at 0.02 person-rem; the collective dose to the public would be 0.14 person-rem. Accordingly, incident-free transportation of radioactive material would result in 7.9×10^{-6} latent fatal cancers among transportation workers, and 0.000072 latent fatal cancers in the total affected population over the duration of the transportation activities.

Impacts of Onsite Accidents During Ground Transportation. The maximum foreseeable onsite transportation accident under this alternative (probability of occurrence: more than 1×10^{-7} per year) would not breech the transportation cask. The probability of more severe accidents also was evaluated, and estimated to be less than 1×10^{-7} per year.

Estimates of the total ground transportation accident risks under this alternative are as follows: a collective dose to the affected population of less than 1×10^{-6} person-rem would result in less than 1×10^{-9} latent fatal cancers; and traffic accidents would result in 0.00011 traffic fatalities.

4.9.6 On- and Offsite Transportation Impacts - Alternative 5

In addition to the 43 shipments of EBR-II driver spent nuclear fuel, Alternative 5 requires 11 shipments of declad EBR-II blanket material and 18 shipments of Fermi-1 blanket material from ANL-W to SRS. Twenty-seven ceramic waste form and 2 metallic waste form shipments would be made from ANL-W to the INEEL Dry Transfer Facility. The impacts for these alternatives include both on- and offsite transportation. The total distance traveled on public roads by trucks carrying radioactive materials would be 111,800 kilometers (69,500 miles).

Impacts of On- and Offsite Incident-Free Transportation. The dose to transportation workers from all transportation activities required by this alternative was estimated at 0.0052 person-rem; the collective dose to the public would be 0.042 person-rem. Accordingly, incident-free transportation of radioactive material would result in 2.1×10^{-6} latent fatal cancers among transportation workers and 0.000021 latent fatal cancers in the total affected population over the duration of the transportation activities. The dose to transportation workers from transporting cleaned and declad blanket spent nuclear fuel to SRS was estimated at 0.0012 person-rem; the collective dose to the public would be 0.012 person-rem. Accordingly, incident-free transportation of radioactive material would result in 4.7×10^{-7} latent fatal cancers among transportation workers and 6×10^{-6} latent fatal cancers in the total affected population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this alternative would be 0.00039.

Impacts of On- and Offsite Accidents During Ground Transportation. The maximum foreseeable offsite transportation accident under this alternative (probability of occurrence: more than 1×10^{-7} per year) would be shipment of EBR-II blanket material from DOE's facility at ANL-W to SRS with a Severity Category 5 accident in a suburban population zone under neutral (average) weather conditions. The accident could result in a dose of 0.00024 person-rem to the public with an associated 1.2×10^{-7} latent fatal cancers, and 2.5×10^{-13} rem to the hypothetical maximally exposed individual with a latent fatal cancer risk of 1.3×10^{-15}. No fatalities would be expected to occur. The probabilities of more severe accidents, different weather conditions at the time of accident, or occurrence in a more densely populated area also were evaluated, and were estimated to be less than 1×10^{-7} per year.

Estimates of the total ground transportation accident risks under this alternative are as follows: a collective dose to the affected population of 3.4×10^{-6} person-rem would result in 1.7×10^{-9} latent fatal cancers; and traffic accidents would result in 0.0018 traffic fatalities. Ground transportation accident risks to the affected population from transporting blanket fuel to SRS were estimated at 3×10^{-6} person-rem, resulting in 1.5×10^{-9} latent fatal cancers; and traffic accidents would result in 0.0017 traffic fatalities.

4.9.7 Onsite Transportation Impacts - Alternative 6

In addition to the 43 shipments of EBR-II driver spent nuclear fuel, 32 ceramic waste form, 1 metallic waste form, and 164 melt and dilute product shipments would be made from ANL-W to the INEEL Dry Transfer

Facility. The total distance traveled on public roads on the INEEL site by trucks carrying radioactive materials would be 9,300 kilometers (5,800 miles).

Impacts of Onsite Incident-Free Transportation. The dose to transportation workers from all transportation activities required by this alternative was estimated at 0.027 person-rem; the collective dose to the public would be 0.20 person-rem. Accordingly, incident-free transportation of radioactive material would result in 0.000011 latent fatal cancers among transportation workers and 0.0001 latent fatal cancers in the total affected population over the duration of the transportation activities.

Impacts of Onsite Accidents During Ground Transportation. The maximum foreseeable onsite transportation accident under this alternative (probability of occurrence: more than 1×10^{-7} per year) would not breech the transportation cask. The probability of more severe accidents also was evaluated, and was estimated to be less than 1×10^{-7} per year.

Estimates of the total ground transportation accident risks under this alternative are as follows: a collective dose to the affected population of less than 1×10^{-6} person-rem would result in less than 1×10^{-9} latent fatal cancers; and traffic accidents would result in 0.00014 traffic fatalities.

4.10 Preferred Alternative

DOE has identified electrometallurgical treatment (Alternative 1) as its Preferred Alternative for the treatment and management of all sodium-bonded spent nuclear fuel, except for the Fermi-1 blanket fuel. The No Action Alternative is preferred for the Fermi-1 blanket spent nuclear fuel; therefore, Fermi-1 blanket spent nuclear fuel would remain in storage, pending a subsequent decision on its long-term management. While EBR-II spent nuclear fuel is undergoing electrometallurgical treatment, DOE has approximately four years[1] in which to evaluate the operating experience of electrometallurgical treatment technology and continue to investigate alternative treatment techniques that currently require additional development for the Fermi-1 blanket spent nuclear fuel. After this data is evaluated, DOE would decide whether to treat the Fermi-1 blanket spent nuclear fuel using electrometallurgical treatment or to use another treatment method and/or disposal technique.

The environmental impacts of the Preferred Alternative, as identified above, are provided in detail in Section 4.2 for the No Action Alternative and in Section 4.3 for Alternative 1. The evaluations provided in these sections cover treatment of both Fermi-1 blanket and other sodium-bonded spent nuclear fuel. The environmental impact contributions from treatment of the Fermi-1 blanket spent nuclear fuel as compared to the EBR-II sodium-bonded spent nuclear are negligible for all resources except for waste management. Overall, the environmental impacts of Alternative 1 bound those of the Preferred Alternative for all resources except for waste management, where the No Action Alternative bounds. The decision to electrometallurgically treat all sodium-bonded spent nuclear fuel except the Fermi-1 blanket fuel would reduce the treatment duration under Alternative 1 from 13 to 7 years. Storing Fermi-1 blanket spent nuclear fuel pending a subsequent decision on its long-term management would not change the duration of the No Action Alternative, i.e., it would remain 35 years.

Should DOE decide to treat Fermi-1 blanket fuel using a treatment method or process, other than electrometallurgical treatment, that was analyzed in Sections 4.4 through 4.7 of this EIS, the environmental consequences would be equal to or bounded by the EIS. As indicated in these sections, all the alternatives analyzed would result in very small and essentially indistinguishable impacts to public and occupational health and safety, air quality, water resources, environmental justice, and transportation. The volumes of waste generated by separate treatment of Fermi-1 blanket fuel would be equal to or bounded by the values presented

[1] *Even though it would take six years to electrometallurgically treat EBR-II spent nuclear fuel at ANL-W, for planning purposes, DOE would need to make the decision in four years.*

for each of the alternatives analyzed in detail. A decision by DOE to treat some or all of the sodium-bonded blanket fuel using a method which has not been analyzed in detail in this EIS would require an evaluation of associated environmental impacts under a separate NEPA document.

4.11 CUMULATIVE IMPACTS

The Council on Environmental Quality regulations implementing NEPA procedural provisions define cumulative impacts as the impacts on the environment which result from the incremental impact of the action when added to other past, present, and reasonably foreseeable future actions, regardless of what agency (Federal or nonfederal) or person undertakes such other actions (40 CFR 1508.7). The cumulative impacts analysis presented in this section is based on the incremental contribution from the maximum impacts from the proposed action added to the baseline conditions at ANL-W and SRS, as well as the maximum impacts from other on- and offsite past, present, and other reasonably foreseeable future actions. Although it is unlikely that the alternative with the maximum impacts would be implemented to treat and manage sodium-bonded spent nuclear fuel at ANL-W and SRS, it was used to estimate cumulative impacts to ensure a conservative analysis. In accordance with a handbook recently prepared by the Council on Environmental Quality, DOE identified the resource areas in which the treatment and management of sodium-bonded spent nuclear fuel could add to the impacts of past, present, and reasonably foreseeable actions within the project impact zones, as defined by the Council on Environmental Quality (CEQ 1997).

Based on an examination of the environmental impacts of the proposed action, coupled with DOE and other agency actions, it was determined that cumulative impacts for the following areas need to be presented: (1) air resources, (2) water resources, (3) socioeconomics, (4) public and worker health, (5) environmental justice, and (6) waste generation. Discussions of cumulative impacts for land resources, site infrastructure, geologic resources, ecological resources, and cultural and paleontological resources were omitted because the related impacts from the proposed treatment and management of sodium-bonded spent nuclear fuel would either not occur or be so small that their potential contribution to cumulative impacts would be negligible.

For determining the impacts to air, water, socioeconomic, human health, environmental justice, and waste generation resources from commercial and Federal nuclear facilities, the 80-kilometer (50-mile) radius surrounding ANL-W and SRS was selected as the project impact zone. For liquid releases from SRS, the downstream population that uses the Savannah River as its source of drinking water was included in the SRS project impact zone.

Cumulative transportation impacts are discussed at the end of this section.

4.11.1 ANL-W and INEEL

Significant offsite activities within a 80-kilometer (50-mile) radius of ANL-W and INEEL that potentially would contribute to the cumulative environmental impacts presented in this analysis include the System Integration Corporation quartzite mining operation in Arco Hills and the Food, Machinery, and Chemical Corporation, a phosphate processing operations in Pocatello, Idaho. The Food, Machinery, and Chemical Corporation is a primary source for offsite radiological emissions. These emissions have been evaluated by the EPA. Radiological impacts from the operation of the phosphate processing operations are minimal, and are not included in assessments at INEEL (DOE 1999a).

The counties surrounding ANL-W and INEEL have a number of existing and planned industrial and commercial facilities with permitted air emissions and water usage. Because of the distances between ANL-W and INEEL and the private industrial facilities, there is little opportunity for the interaction of plant emissions and no significant contribution to the cumulative impact on air or water. Reasonably foreseeable offsite actions evaluated in this EIS are presented in **Table 4–64**.

Table 4–64 Offsite Activities Included in the Assessment of Cumulative Impacts at ANL-W and INEEL

Activity	*Description*
Housing development, Idaho Falls	300-unit single family housing development planned on approximately 150 acres of vacant land.
Business park, Rexburg	50 acres of vacant land between two light industrial facilities planned for expansion into a light industrial/business park for 30-40 businesses.
Manufacturer, Pocatello	Existing manufactured-home factory to expand from approximately 50 to between 140 and 150 employees. Expansion of 22 acres in Pocatello Airport Industrial Park.
Food, Machinery, and Chemical Corp., Pocatello	Phosphate manufacturing plant to reduce number of furnaces from four to three within the next two years; 25-30 jobs could be lost.
System Integration Corporation Arco Hills Quartzite Mine	Quartzite mining operation and ore processing near Arco Hills on 56 acres. Fourteen acres would be disturbed by the quarry operation and a small waste ore dump; 22 acres would be disturbed by the construction of a haul road; 11 acres would be disturbed by the ore crushing facilities; and 9 acres would be disturbed by the loading facilities at INEEL. The project would employ 40 workers.

Source: DOE 1999a.

The cumulative impacts analysis also addressed the contributory effects from other past, present, and reasonably foreseeable future DOE actions at ANL-W and INEEL. These actions and their associated NEPA documentation are summarized in Section 1.6. The contributory effects of impacts from actions proposed in the *Idaho High-Level Waste and Facilities Disposition Environmental Impact Statement* (DOE 1999e) were included in the cumulative impact analysis. In this EIS, DOE evaluated the impacts from the proposed construction and operation of a high-level radioactive waste and liquid sodium-bearing waste treatment facility at INEEL to make these materials ready for disposal. This project also involves the disposition of high-level radioactive waste generation, storage, and treatment facilities at INEEL upon the completion of their missions.

Other reasonably foreseeable future actions that may contribute to cumulative impacts at INEEL but were not included in this analysis include a proposed DOE Office of Nuclear Energy, Science, and Technology project. This project involves evaluating INEEL as a potential site for the production of plutonium-238 for use in radioisotope power systems for future space missions. This project would include the use of INEEL's Fluorinel Dissolution Process Facility at INTEC for either storing neptunium-237 and/or fabricating and processing neptunium-237 targets to produce plutonium-238, and the use of the Advanced Test Reactor for the irradiation of neptunium-237 targets. The Advanced Test Reactor is an operating test reactor with a programmatic mission to support the Naval Reactor Fuels Program. Public scoping for this project has been completed. A preliminary review of the project indicates that there would be a contributory effect to the cumulative impacts—primarily to public and worker health and safety due to the fabrication and processing of neptunium-237 targets in the Fluorinel Dissolution Process Facility, loading and unloading targets in the Advanced Test Reactor, and handling of irradiated targets for packaging and shipping. The cumulative impacts from this proposed project will be addressed in a separate NEPA document.

The proposed commercial project (VentureStar) would involve a commercial spin-off of the National Aeronautics and Space Administration's Reusable Launch Vehicle research program that would replace the existing Space Shuttle Program. INEEL is being considered as a potential candidate site for both the launch and landing of this next-generation spacecraft. The project is in the very early stages of development and does not appear to be near term (5 to 10 years). Cumulative impacts from this proposed project would be addressed in separate NEPA documentation.

The cumulative impacts analysis also included the impacts from actions proposed in this EIS. Risks to members of the public and site workers from radiological and nonradiological releases were based on operational impacts from the alternatives described in Chapter 4 of this EIS.

Temporal limits were defined by examining the period of influence from both the proposed action and other Federal and nonfederal actions that have the potential for cumulative impacts. Actions to support the treatment of sodium-bonded blanket spent nuclear fuel at ANL-W are expected to begin in 2000 in preparation for ultimate offsite disposal, possibly in a geologic repository which probably will not be available until at least 2010. Final offsite shipments of spent nuclear fuel at ANL-W and INEEL for disposal would be completed by 2035.

The period of interest for the cumulative impacts analysis for this EIS includes the proposed construction, operation, and disposition of facilities identified in the *Idaho High-Level Waste and Facilities Disposition Environmental Impact Statement* (DOE 1999e) and the *Advanced Mixed Waste Treatment Project Final Environmental Impact Statement* (DOE 1999a), while actions for other nuclear materials and surplus plutonium disposition would be ongoing.

4.11.1.1 Air Resources

Table 4–65 compares the cumulative concentrations of nonradiological air pollutants from INEEL to Federal and state regulatory standards. The listed values are the maximum modeled concentrations that could occur at the ground level at the site boundary. The data demonstrate that the total estimated concentrations of nonradiological air pollutants from INEEL in the all cases would be well below the regulatory standards at the site boundary. Among the pollutants, the concentrations of nitrogen oxides come closest to the standard (14 percent of the standard for the annual averaging time). The remaining pollutant emissions would result in concentrations below 13 percent of the applicable standards. As indicated in this table, the values presented in the INEEL baseline include concentrations from releases at ANL-W. ANL-W's criteria pollutant concentrations are from currently operating equipment, which are not expected to increase under any of the alternatives. Therefore, there would be no contribution to cumulative air quality impacts at INEEL as a result of the proposed action.

4.11.1.2 Water Resources

There would be no liquid effluent released to surface water or groundwater from the operation of ANL-W or INEEL facilities as a result of the proposed action. Therefore, there would be no contribution to the cumulative impact.

4.11.1.3 Socioeconomic Impacts

No additional workers would be required for the operation of ANL-W or INEEL facilities as a result of the proposed action. Therefore, there would be no contribution to the cumulative impact.

4.11.1.4 Public and Worker Health

Table 4–66 summarizes the cumulative radiological health effects of routine ANL-W and INEEL operations, proposed DOE actions, and nonfederal nuclear facility operations. Impacts resulting from proposed DOE actions are described in the various EISs listed in Section 1.6. In addition to estimated radiological doses to the maximally exposed offsite individual, the offsite population, and workers, Table 4–66 lists the potential number of latent cancer fatalities for the public and workers from exposure to radiation. The radiation dose to the maximally exposed offsite individual would be 0.047 millirem per year, which is well below the applicable DOE regulatory limits (10 millirem per year from the air pathway [40 CFR 61] and 100 millirem

per year for all pathways). The total annual population dose of 0.35 person-rem for current and projected activities translates into 0.00017 latent cancer fatalities for each year of exposure for the population living within a 80-kilometer (50-mile) radius of the ANL-W.

Table 4–65 Estimated Maximum Cumulative Ground-Level Concentrations of Nonradiological Pollutants (micrograms per cubic meter) at the INEEL Boundary

Pollutant	*Averaging Time*	*Most Stringent Standard or Guideline* [a]	*INEEL Baseline* [b]	*Advanced Mixed Waste Treatment Project EIS* [c]	*Idaho High-Level Waste and Facilities Disposition EIS* [d]	*Cumulative Concentrations* [e]
Carbon monoxide	8 hours	10,000 [f]	120	1	4.2	130
	1 hour	40,000 [f]	265	115	10	390
Nitrogen oxides	Annual	100 [f]	13	0.3	0.2	14
PM_{10}	Annual	50 [f]	0.65	0.006	0.02	1
	24 hours	150 [f]	13	4.6	0.3	18
	24 hours (99^{th} percentile over 3 years)	150 [g]	Not available	Not available	Not available	Not available
$PM_{2.5}$	3-year annual	15 [g]	Not available	Not available	Not available	Not available
	24 hours (98^{th} percentile over 3 years)	65 [g]	Not available	Not available	Not available	Not available
Sulfur dioxide	Annual	80 [f]	3.4	0.012	0.57	4
	24 hours	365 [f]	32	4.5	9	46
	3 hours	1,300 [f]	84	25	42	151

PM_n = Particulate matter less than or equal to n microns in diameter.

[a] The more stringent Federal or state standard is presented if both exist for the averaging period.

[b] INEEL baseline includes concentrations from releases at ANL-W which, in turn, include releases under all alternatives considered in the SBSNF EIS (see Section 3.2.3).

[c] DOE 1999a: Table 5.7-6, Preferred Alternative (Microencapsulation option).

[d] DOE 1999e: Table C.2-14, Separation (Planning Basis) option.

[e] Values presented in this column could be different from the sum of the individual values due to rounding.

[f] Federal and state standard.

[g] Federal standard.

The annual collective dose to the worker population would be 200 person-rem. In addition, doses to individual workers would be kept below the DOE Administrative Control Level of 2,000 millirem per year, as established for all DOE activities in DOE Order N 441.1, which is well below the regulatory limit of 5,000 millirem per year (10 CFR 835). Furthermore, "as low as reasonably achievable" principles would be exercised to maintain individual worker doses below the DOE Administrative Control Level of 2,000 millirem per year.

4.11.1.5 Environmental Justice

As discussed in Chapter 4 and Appendix H, implementation of the alternatives for the treatment and management of sodium-bonded spent nuclear fuel at ANL-W and INEEL would have no significant impact on public health or the environment. Therefore, the implementation of the proposed action or the No Action Alternative would result in no disproportionately high and adverse impacts on minority or low-income populations residing within potentially affected areas.

Table 4–66 Estimated Annual Cumulative Radiological Doses and Resulting Health Effects to Offsite Population and Facility Workers at ANL-W and INEEL

Activity	Maximally Exposed Offsite Individual		Population [a]		Workers	
	Dose (Millirem)	Latent Cancer Fatality Risk	Collective Dose (Person-Rem)	Excess Latent Cancer Fatalities	Collective Dose (Person-Rem)	Excess Latent Cancer Fatalities
ANL-W and INEEL Baseline [b]	0.021	1.1×10^{-8}	0.23	0.00012	115	0.046
SBSNF EIS [c]	0.002	1.0×10^{-9}	0.012	6.0×10^{-6}	22	0.0088
Advanced Mixed Waste Treatment Program [d]	0.022	1.1×10^{-8}	0.009	4.5×10^{-6}	4.1 [e]	0.0016
High-Level Waste and Facilities Disposition [f]	0.002	1.0×10^{-9}	0.10	0.00005	59	0.024
Total	0.047	2.4×10^{-8}	0.35	0.0017	200	0.08

[a] A collective dose to the 80-kilometer (50-mile) population from atmospheric releases. There would be no liquid releases from ANL-W and INEEL facilities as a result of the proposed action.
[b] Data from Tables 3–9 and 3–10 of this SBSNF EIS.
[c] Alternative 6: Melt and dilute blanket and driver fuel at ANL-W.
[d] DOE 1999a: Tables 5.12–1 and E.4–7. Preferred Alternative (Microencapsulation Option) Record of Decision (64 FR 16948).
[e] Estimate based on the number of workers and the average dose per worker, i.e., 50 workers (DOE 1999a: Table E.4–7) × 81 millirem (DOE 1999a: Table 5.12–1) = 4050 person millirem = 4.1 person-rem.
[f] DOE 1999e: Table 5.4–6, maximum dose for any alternative. Average annual dose of 190 millirem per worker.

4.11.1.6 Waste Generation

As stated in the Waste Management discussions for each alternative presented earlier in Chapter 4, low-level radioactive waste, mixed and hazardous waste, and transuranic waste would be generated by the treatment of sodium-bonded spent nuclear fuel. Under the proposed action (with the partial exception of Alternative 2), the existing sodium-bonded spent nuclear fuel inventories at ANL-W and INEEL would be converted into a high-level radioactive waste form for disposal in a geologic repository and, therefore, the volume of the high-level radioactive waste that would be generated is not counted as new waste—"high-level radioactive waste." In fact, under the proposed action, the amount of material at ANL-W and INEEL scheduled for disposal in a geologic repository would decrease. For all alternatives under the proposed action, the volume of the new high-level radioactive waste forms would be less than the volume of untreated sodium-bonded spent nuclear fuel (the No Action Alternative). However, as stated in the Waste Management discussions, the projected amount of high-level radioactive waste would not require additional treatment and storage capacities beyond the current and planned INEEL capacities.

Table 4–67 lists the cumulative total waste generated at ANL-W and INEEL for years 2000 to 2035. The estimated quantity of radioactive/hazardous waste from baseline operations in this forecast through the year 2035 would be 205,550 cubic meters (7.25 million cubic feet). Waste generated by Alternative 6: Melt and dilute blanket and driver fuel at ANL-W (the alternative generating the most waste in all categories) would add a total of 980 cubic meters (34,610 cubic feet). During a 15-year time period, other reasonably foreseeable activities associated with the treatment of high-level radioactive waste and facility disposition at INEEL could add an additional 30,730 cubic meters (1.1 million cubic feet). Therefore, the potential cumulative total amount of waste generated from ANL-W and INEEL activities would be 237,260 cubic meters (8.4 million cubic feet).

Table 4–67 Estimated Cumulative Total Waste Generation for Years 2000 to 2035 From ANL-W and INEEL Concurrent Activities (Cubic Meters)

Waste Type	*ANL-W and INEEL Baseline Operations* [a]	*Idaho HLW and Facility Disposition EIS* [a]	*SBSNF EIS* [b]	*Total*
High-level radioactive	0	0	0 [d]	0 [c]
Low-level radioactive	135,600	15,320	925	151,845
Hazardous/mixed low-level radioactive	4,950	15,300	40	20,290
Transuranic	65,000 [d]	110	15	65,125
Total	205,550	30,730	980	237,260

HLW = High-level radioactive waste.

[a] DOE 1999e: Figures 5.4–1 through 5.4–3 and input values for those figures through year 2035, Separations Alternative. Maximum quantities for any alternative.

[b] Alternative 6: Melt and dilute blanket and driver fuel at ANL-W; 12 years of operation. This alternative would generate the most waste in all categories.

[c] During treatment, the sodium-bonded spent nuclear fuel from existing inventories at ANL-W and INEEL would be converted into a high-level radioactive waste form for disposal in a geologic repository. For any alternative, the amount of material at ANL-W and INEEL scheduled for disposal in a geologic repository would not increase.

[d] In storage at the Radioactive Waste Management Complex.

The Central Facilities Area and Bonneville County landfill accepts nonhazardous and nonradioactive solid waste generated at INEEL. The onsite landfill complex was designed to accommodate combined ANL-W and INEEL solid waste disposal needs for a projected maximum operational life of 30 years.

The activities supporting the treatment and management of sodium-bonded spent nuclear fuel and other planned ANL-W and INEEL activities would not generate larger volumes of radioactive, hazardous, or solid waste beyond the current and projected capacities of ANL-W and INEEL waste storage and/or management facilities.

4.11.2 Savannah River Site

Nuclear facilities within a 80-kilometer (50-mile) radius of SRS include Georgia Power's Vogtle Electric Generating Plant across the river from SRS; Chem-Nuclear Inc., a commercial low-level radioactive waste burial site just east of SRS; and Starmet CMI, Inc. (formerly Carolina Metals), located southeast of SRS, which processes uranium-contaminated metals. Radiological impacts from the operation of the Vogtle Electric Generating Plant, a two-unit commercial nuclear power plant, are minimal, but DOE has factored them into the analysis. As stated in the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000), the South Carolina Department of Health and Environmental Control Annual Report indicates that operation of the Chem-Nuclear Services facility and the Starmet CMI facility do not noticeably impact radiation levels in air or liquid pathways in the vicinity of SRS. Therefore, they are not included in this assessment.

The counties surrounding SRS have numerous existing (e.g., textile mills, paper product mills, and manufacturing facilities) and planned (e.g., Bridgestone Tire) industrial facilities with permitted air emissions and discharges to surface waters. Because of the distances between SRS and the private industrial facilities, there is little opportunity for interactions of plant emissions, and no major cumulative impact on air or water quality. Construction and operation of Bridgestone Tire and Hankook Polyester facilities could affect the regional socioeconomic cumulative impacts.

Additional offsite facilities with the potential to affect the nonradiological environment include South Carolina Electric and Gas Company's Urquhart Station. Urquhart Station is a three-unit, 250-megawatt, coal- and natural gas-fired steam electric plant on Beech Island, South Carolina, located north of SRS. Because of the

distance between SRS and the Urquhart Station and the regional wind direction frequencies, there is little opportunity for any interaction of plant emissions, and no significant cumulative impact on air quality.

DOE also evaluated the impacts from its own proposed future actions by examining impacts to resources and the human environment, as shown in NEPA documentation related to SRS (see Section 1.6). Additional NEPA documents related to SRS that are considered in the cumulative impacts section include the following:

Environmental Assessment for the Tritium Facility Modernization and Consolidation Project at the Savannah River Site (DOE 1998a). This environmental assessment addresses the impacts of consolidating the tritium activities currently performed in Building 232-H into the new Building 233-H and Building 234-H. Tritium extraction functions would be transferred to the Tritium Extraction Facility. The overall impact would be to reduce the tritium facility complex net tritium emissions by up to 50 percent. Another positive effect of this planned action would be to reduce the amount of low-level radioactive job-control waste. Effects on other resources would be negligible. Therefore, impacts from the environmental assessment have not been included in this cumulative impacts analysis.

Final Environmental Impact Statement on Management of Certain Plutonium Residues and Scrub Alloy Stored at the Rocky Flats Environmental Technology Site (DOE 1998d). DOE proposes to process certain plutonium-bearing materials being stored at the Rocky Flats Environmental Technology Site. These materials are plutonium residues and scrub alloy remaining from nuclear weapons manufacturing operations formerly conducted by DOE at Rocky Flats. DOE has decided to remove the plutonium from certain residues that would be shipped from the Rocky Flats Environmental Technology Site to SRS for stabilization. The separated plutonium would be stored at SRS pending disposition decisions. Environmental impacts from using the F-Canyon to chemically separate the plutonium from the remaining materials at SRS are included in this section.

Draft and Final Environmental Impact Statements for the Construction and Operation of a Tritium Extraction Facility at the Savannah River Site (DOE 1998b, DOE 1999b). DOE proposes to construct and operate a Tritium Extraction Facility at SRS to provide the capability to extract tritium from commercial light water reactor targets and targets of similar design. The purpose of the proposed action and alternatives evaluated in the EIS is to provide tritium extraction capability to support reactor tritium production. Environmental impacts from the maximum processing option in this EIS are included in this section.

Defense Waste Processing Facility Supplemental Environmental Impact Statement (DOE 1994). The selected alternative in the Record of Decision was the completion and operation of the Defense Waste Processing Facility to immobilize high-level radioactive waste at SRS. The facility is currently processing sludge from SRS high-level radioactive waste tanks. However, SRS baseline data are not representative of full Defense Waste Processing Facility operational impacts, including the processing of salt and supernate from these tanks. Therefore, the Defense Waste Processing Facility data is listed separately.

In addition, the cumulative impacts analysis also includes the impacts from actions proposed in this SBSNF EIS. Risks to members of the public and site workers from radiological and nonradiological releases are based on operational impacts from the alternatives described in Sections 4.5 and 4.7, of this EIS.

Temporal limits were defined by examining the period of influence from both the proposed action and other Federal and nonfederal actions that have the potential for cumulative impacts. Actions to support the treatment of sodium-bonded blanket spent nuclear fuel at SRS are expected to begin in 2003 in preparation for ultimate offsite disposal, possibly in a geologic repository which probably will not be available until at least 2010. Final offsite shipments for spent nuclear fuel currently assigned to SRS for disposal would be completed by 2035.

The period of interest for the cumulative impacts analysis for this EIS includes the proposed construction and operation of facilities identified in the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000) and the Draft EIS for the SRS Tritium Extraction Facility (DOE 1998b, DOE 1999b), while actions for nuclear materials, highly enriched uranium, and surplus plutonium disposition would be ongoing.

4.11.2.1 Air Resources

Table 4–68 compares the cumulative concentrations of nonradiological air pollutants from SRS to Federal and state regulatory standards. The listed values are the maximum modeled concentrations that could occur at ground level at the site boundary. The data demonstrate that total estimated concentrations of nonradiological air pollutants from SRS in all cases would be below the regulatory standards at the site boundary. Among the pollutants, the concentration of sulfur dioxide comes closest to the standard (approximately 96 percent of the standard for the 24-hour averaging time). The remaining pollutant emissions would range from 25 to 93 percent of the applicable standards.

Table 4–68 Estimated Maximum Cumulative Ground-Level Concentrations of Nonradiological Criteria Pollutants (Micrograms per Cubic Meter) at the SRS Boundary

Pollutant	*Averaging Time*	*Most Stringent Standard or Guideline* [a]	*SRS Baseline* [b]	*SBSNF EIS* [c]	*Other Foreseeable Planned SRS Activities* [d]	*Cumulative Concentrations* [e]
Carbon monoxide	8 hours	10,000 [f]	6,900	1.22	6.78	6,908
	1 hour	40,000 [f]	10,000	9.06	44.63	10,054
Nitrogen dioxide	Annual	100 [f]	26	3.11	4.63	34
PM_{10}	Annual	50 [f]	25	Less than 0.01	0.21	25
	24 hours (interim)	150 [f]	130	0.11	6.82	137
	24 hours (99th percentile over 3 years)	150 [g]	(h)	Not available	Not available	Not available
$PM_{2.5}$	3-year annual	15 [g]	(h)	Not available	Not available	Not available
	24 hours (98th percentile over 3 years)	65 [g]	(h)	Not available	Not available	Not available
Sulfur dioxide	Annual	80 [f]	34	Less than 0.01	0.06	34
	24 hours	365 [f]	350	0.12	0.96	351
	3 hours	1,300 [f]	1,200	0.91	5.28	1,206

PM_n = Particulate matter less than or equal to n microns in diameter.
[a] The more stringent Federal or state standard is presented if both exist for the averaging period.
[b] Data from Table 3–16 of this EIS.
[c] Alternative 3: PUREX process blanket fuel at SRS F-Canyon.
[d] Data compiled from SRS Spent Nuclear Fuel Management Final EIS (DOE 2000: Table 5–1), including contributions from the Preferred Alternative less contributions from SBSNF EIS.
[e] Values in this column are rounded to the nearest number.
[f] Federal and state standard.
[g] Federal standard.
[h] No data available with which to assess particulate matter concentrations.

DOE also evaluated the cumulative impacts of airborne radiological releases in terms of dose to a maximally exposed offsite individual at the SRS boundary. DOE included the impacts of the Vogtle Plant (NRC 1996) in this cumulative total. The radiological emissions from the operation of the Chem-Nuclear low-level radioactive waste disposal facility just east of SRS are very low (DOE 2000) and are not included. **Table 4–69** lists the results of this analysis, using 1996 emissions (1992 for the Vogtle Plant) for the SRS baseline. The cumulative dose to the maximally exposed offsite individual would be 0.10 millirem per year, well below the regulatory standard of 10 millirem per year (40 CFR Part 61). Summing the doses to the maximally exposed

offsite individual for the proposed action and baseline SRS operations listed in Table 4–69 is an extremely conservative approach because, to get the calculated dose, the maximally exposed offsite individual would have to occupy different physical locations at the same time, which is impossible.

Table 4–69 Estimated Average Annual Cumulative Radiological Doses and Resulting Health Effects to the Offsite Population in the 80-Kilometer (50-Mile) Radius From Airborne Releases at SRS

Activity	Maximally Exposed Offsite Individual		Population [a]	
	Dose (millirem)	Latent Cancer Fatality Risk	Collective Dose (person-rem)	Excess Latent Cancer Fatalities
SRS baseline [b]	0.050	2.5×10^{-8}	5.5	0.0028
SBSNF EIS [c]	0.00039	2.0×10^{-10}	0.019	9.5×10^{-6}
Management of spent nuclear fuel [d]	0.015	7.5×10^{-9}	0.56	0.00028
Disposition of surplus highly enriched uranium [d]	0.0025	1.3×10^{-9}	0.16	0.00008
Tritium Extraction Facility [d]	0.02	1.0×10^{-8}	0.77	0.00039
Disposition of surplus plutonium [d]	0.0074	3.7×10^{-9}	1.8	0.0009
Management of plutonium residues/scrub alloy [d]	0.00057	2.9×10^{-10}	0.0062	3.1×10^{-6}
Defense Waste Processing Facility [d]	0.001	5.0×10^{-10}	0.071	0.000036
DOE Complex miscellaneous components [d]	0.0044	2.2×10^{-9}	0.007	3.3×10^{-6}
Vogtle Plant [d]	0.00054	2.7×10^{-10}	0.042	0.000021
Total	0.10	5.1×10^{-8}	8.94	0.0045

[a] A collective dose to the 80-kilometer (50-mile) population.
[b] Data from Table 3–21 of this EIS.
[c] Alternative 3: PUREX process blanket fuel at SRS F-Canyon.
[d] Data from SRS Spent Nuclear Fuel Management Final EIS (DOE 2000: Table 5–2 maximum impact alternative).

Adding the population doses from current and projected activities at SRS, the Vogtle Plant, the SRS Spent Nuclear Fuel Management Final EIS, and this EIS could yield a total annual cumulative dose of 8.94 person-rem from airborne sources. The total annual cumulative dose translates into 0.0045 latent cancer fatalities for each year of exposure for the population living within an 80-kilometer (50-mile) radius of SRS.

4.11.2.2 Water Resources

At present, a number of SRS facilities discharge treated wastewater to Upper Three Runs and its tributaries and Fourmile Branch via NPDES–permitted outfalls. These include the F- and H-Area Effluent Treatment Facility and the M-Area Liquid Effluent Treatment Facility. As stated in Sections 4.5.2 and 4.7.2, operations associated with the treatment and management of sodium-bonded spent nuclear fuel are not expected to result in any discharges to groundwater. The only technology that would result in discharges of radiological and nonradiological effluent to surface water would be PUREX processing. The major sources of liquid effluent from facilities associated with PUREX processing would be process cooling water and steam condensate systems that could contain small quantities of radionuclides and chemicals. This process wastewater would be treated at the F-Area Effluent Treatment Facility and then discharged to the Upper Three Runs. Studies of water quality and biota downstream of the Effluent Treatment Facility outfall suggest that discharges have not degraded the water quality of Upper Three Runs (DOE 2000).

Other potential sources of contaminants into Upper Three Runs during the periods in which sodium-bonded spent nuclear fuel would be treated in F-Area using PUREX, or in L-Area using melt and dilute treatment, include activities described in the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000), the tritium extraction facility, environmental restoration, and decontamination and deactivation activities, as well as modifications to existing SRS facilities. Discharges from activities associated with the SRS Spent Nuclear Fuel Management Final EIS and tritium extraction facility would not add significant amounts of nonradiological contaminants to Upper Three Runs. The amount of discharge associated with environmental restoration and decontamination and deactivation activities would vary based on the level of activity. All the potential activities that could result in wastewater discharges would be required to comply with the NPDES permit limits that ensure protection of water quality.

Table 4–70 summarizes the estimated cumulative radiological doses from waterborne sources to human receptors downstream from SRS. Liquid effluent released to SRS streams that are tributaries of the Savannah River could contain small quantities of radionuclides. The exposure pathways considered in this analysis included drinking water, fish ingestion, shoreline exposure, swimming, and boating. The estimated cumulative dose to the maximally exposed offsite individual from liquid releases would be 0.24 millirem per year, well below the regulatory standard of 4 millirem per year (40 CFR Part 141). Adding the population doses associated with current and projected SRS activities would yield a cumulative annual dose of 2.6 person-rem from liquid sources. This translates into 0.0013 latent cancer fatalities for each year of exposure of the population living within an 80-kilometer (50-mile) radius of SRS.

Table 4–70 Estimated Average Annual Cumulative Radiological Doses and Resulting Health Effects to the Offsite Population in the 80-Kilometer (50-Mile) Radius From Liquid Releases at SRS

	Maximally Exposed Offsite Individual		*Population* [a]	
Activity	*Dose (millirem)*	*Latent Cancer Fatality Risk*	*Collective Dose (person-rem)*	*Excess Latent Cancer Fatalities*
SRS baseline [b]	0.13	6.5×10^{-8}	2.4	0.0012
SBSNF EIS [c]	0.00012	6.0×10^{-11}	0.00068	3.4×10^{-7}
Management of spent nuclear fuel [d]	0.057	2.9×10^{-8}	0.19	0.000095
Disposition of surplus highly enriched uranium [d]	(e)	(e)	(e)	(e)
Tritium Extraction Facility [d]	(e)	(e)	(e)	(e)
Defense Waste Processing Facility [d]	(e)	(e)	(e)	(e)
Disposition of surplus plutonium [d]	(e)	(e)	(e)	(e)
Management plutonium residues/scrub alloy [d]	(e)	(e)	(e)	(e)
DOE Complex miscellaneous components [d]	0.000042	2.1×10^{-11}	0.00024	1.2×10^{-7}
Plant Vogtle [d]	0.054	2.7×10^{-8}	0.0025	1.3×10^{-6}
Total	0.24	1.2×10^{-7}	2.6	0.0013

[a] A collective dose to the 80-kilometer (50-mile) and downstream population.
[b] Data from Table 3–21 of this EIS.
[c] Alternative 3: PUREX process blanket fuel at SRS F-Canyon.
[d] Data from SRS Spent Nuclear Fuel Management Final EIS (DOE 2000: Table 5-3 maximum impact alternative).
[e] Less than minimum reportable levels.

4.11.2.3 Socioeconomic Impacts

No additional workers would be required for the operation of PUREX and melt and dilute facilities at SRS as a result of the proposed action. Therefore, there would be no contribution to the cumulative impacts.

4.11.2.4 Public and Worker Health

Table 4–71 summarizes the cumulative radiological health effects of routine SRS operations, proposed DOE actions, and nonfederal nuclear facility operations (Vogtle Electric Generating Plant). Impacts resulting from proposed DOE actions are described in the EISs listed previously in this chapter. In addition to estimated radiological doses to the hypothetical maximally exposed offsite individual, the offsite population, and the workers, Table 4–71 lists the potential number of latent cancer fatalities for the public and workers due to radiation exposure. The radiation dose to the maximally exposed offsite individual from air and liquid pathways would be 0.34 millirem per year, which is well below the applicable DOE regulatory limits (10 millirem per year from the air pathway; 4 millirem per year from the liquid pathway; and 100 millirem per year for all pathways). The total annual population dose for current and projected activities of 11.5 person-rem translates into 0.0058 latent cancer fatalities for each year of exposure for the population living within an 80-kilometer (50-mile) radius of SRS.

The annual radiation dose to a worker population would be 859 person-rem. In addition, doses to individual workers would be kept below the DOE Administrative Control Level of 2,000 millirem per year, as established for all DOE activities in DOE Order N 441.1, which is well below the regulatory limit of 5,000 millirem per year (10 CFR 835). Furthermore, standards and practices to ensure worker doses are as low as reasonably achievable would be exercised to maintain individual worker doses below the DOE Administrative Control Level of 2,000 millirem per year.

4.11.2.5 Environmental Justice

As discussed in Chapter 4 and Appendix H, implementation of the alternatives for the treatment and management of sodium-bonded spent nuclear fuel at SRS would have no significant impact on public health or the environment. Therefore, the implementation of either of two alternatives at SRS would result in no disproportionately high and adverse impacts on minority or low-income populations residing within potentially affected areas.

4.11.2.6 Waste Generation

As stated in Sections 4.5.6 and 4.7.6, high-level and low-level radioactive waste, transuranic waste, and hazardous/mixed waste would be generated from the treatment of declad and cleaned sodium-bonded blanket spent nuclear fuel at SRS. The largest volumes of low-level radioactive and transuranic waste would be generated with PUREX processing. However, as stated in Sections 4.5.6 and 4.7.6, the projected waste generation rates would not require additional treatment and storage capacities beyond the current and planned SRS capacities. It should be noted that the treatment of blanket spent nuclear fuel at SRS would result in the generation of new high-level radioactive waste that would be added to the SRS current inventory. This is because the blanket spent nuclear fuel would be transported from ANL-W to SRS for treatment.

Table 4–72 lists the cumulative volumes of liquid high-level and solid low-level radioactive, transuranic, and hazardous/mixed waste that SRS would generate. The table includes data from the SRS 30-year expected waste forecast. The 30-year expected waste forecast is based on operations, environmental restoration, and decontamination and deactivation waste forecasts from existing generators and the following assumptions: (1) secondary waste from the Defense Waste Processing Facility, In-Tank Precipitation, and Extended Sludge Processing operations are addressed in the Defense Waste Processing Facility EIS; (2) high-level radioactive waste volumes are based on the selected option for the F-Canyon Plutonium Solutions EIS; (3) some investigation-derived waste is handled as hazardous waste per RCRA regulations; (4) purge water from well samplings is handled as hazardous waste; and (5) the continued receipt of small amounts of low-level radioactive waste from other DOE facilities and nuclear naval operations (DOE 2000).

Table 4–71 Estimated Average Annual Cumulative Radiological Doses and Resulting Health Effects to the Offsite Population and Facility Workers at SRS

Activity	Maximally Exposed Offsite Individual				Population [a]				Workers	
	Dose From Airborne Releases (millirem)	Dose From Liquid Releases (millirem)	Total Dose (millirem)	Latent Cancer Fatality Risk	Dose From Airborne Releases (person-rem)	Dose From Liquid Releases (person-rem)	Collective Dose (person-rem)	Excess Latent Cancer Fatalities	Collective Dose (person-rem)	Excess Latent Cancer Fatalities
SRS baseline [b]	0.050	0.13	0.18	9.5×10^{-8}	5.5	2.4	7.9	0.0025	165	0.066
SBSNF EIS [c]	0.00039	0.00012	0.00051	2.6×10^{-10}	0.019	0.00068	0.020	1.0×10^{-8}	38	0.015
Management of spent nuclear fuel [d]	0.015	0.057	0.072	3.6×10^{-8}	0.56	0.19	0.75	0.00038	55	0.022
Disposition of surplus highly enriched uranium [d]	0.0025	(e)	0.0025	1.3×10^{-8}	0.16	(e)	0.16	0.00008	11	0.00044
Tritium Extraction Facility [d]	0.02	(e)	0.02	1.0×10^{-8}	0.77	(e)	0.77	0.00039	4	0.0016
Defense Waste Processing Facility [d]	0.001	(e)	0.001	5.0×10^{-10}	0.071	(e)	0.071	0.000036	120	0.048
Disposition of surplus plutonium [d]	0.0074	(e)	0.0074	3.7×10^{-9}	1.8	(e)	1.8	0.0009	456	0.18
Management plutonium residues/ scrub alloy [d]	0.00057	(e)	0.00057	2.9×10^{-10}	0.0062	(e)	0.0062	3.1×10^{-6}	7.6	0.003
DOE Complex miscellaneous components [d]	0.0044	0.000042	0.0044	2.2×10^{-9}	0.007	0.00024	0.0072	3.6×10^{-6}	2	0.001
Vogtle Plant [d]	0.00054	0.054	0.055	2.7×10^{-8}	0.042	0.0025	0.045	0.000022	Not available	
Total	0.10	0.24	0.34	1.9×10^{-7}	8.95	2.60	11.5	0.0058	859	0.34

[a] A collective dose to the 80-kilometer (50-mile) population for atmospheric releases and to the downstream users of the Savannah River for liquid releases.
[b] Data from Tables 3–21 and 3–22 of this EIS.
[c] Alternative 3: PUREX process blanket fuel at SRS F-Canyon.
[d] Data from SRS Spent Nuclear Fuel Management Final EIS (DOE 2000: Table 5-4).
[e] Less than minimum reportable levels.

Table 4–72 Estimated Cumulative Total Waste Generation From SRS Concurrent Activities (Cubic Meters)

Waste Type	*SRS Baseline Operations* [a]	*SBSNF EIS* [b]	*Spent Nuclear Fuel Management* [a]	*ER/D&D* [a]	*Other Waste Volume* [a]	*Total* [c]
Liquid high-level radioactive	14,129	510	11,000	0	69,040	94,680
Low-level radioactive	118,669	900	140,000	61,630	109,200	430,400
Hazardous/mixed low-level radioactive	3,856	7	270	6,173	4,430	14,740
Transuranic	6,012	90	3,700	0	8,730	18,530
Total [c]	142,670	1,510	154,970	67,800	191,400	558,350

ER/D&D = environmental restoration/decontamination and decommissioning; based on a total 30-year expected waste forecast, including previously generated waste.

[a] Data from SRS Spent Nuclear Fuel Management Final EIS (DOE 2000) maximum impact alternative, Table 5-5, based on a total 30-year expected waste forecast, including previously generated waste, and adjusted for the SBSNF EIS.

[b] Alternative 3: PUREX processing of declad and cleaned blanket spent nuclear fuel at SRS F-Canyon.

[c] The values are rounded to the nearest ten; the total sum may be different from the sum of individuals.

As indicated in Table 4–72, the estimated quantity of radioactive/hazardous waste from SRS operations in this forecast during the next 30 years would be approximately 142,670 cubic meters (5.04 million cubic feet). Waste generated by Alternative 3: PUREX processing of blanket fuel at SRS F-Canyon, would add a total of approximately 1,510 cubic meters (53,330 cubic feet). Waste generated from the conventional (PUREX) processing option described in the SRS Spent Nuclear Fuel Management Final EIS would add a total of 154,970 cubic meters (5.48 million cubic feet). In addition, radioactive/hazardous waste associated with environmental restoration and decontamination and decommissioning activities would have a 30-year expected forecast of approximately 67,800 cubic meters (2.39 million cubic feet) (DOE 2000). During this same time period, other reasonably foreseeable activities that were not included in the 30-year forecast would add, approximately, an additional 191,400 cubic meters (6.76 million cubic feet). Therefore, the potential cumulative amount of waste generated from SRS activities during the period of interest would be approximately 558,350 cubic meters (19.72 million cubic feet). It is important to note that the quantities of waste generated are not equivalent to the amounts that will require disposal. At SRS, high-level radioactive material is evaporated and concentrated to a smaller volume for final disposal. Combustible low-level radioactive waste is volume-reduced on site in the Consolidated Incineration Facility.

The Three Rivers Solid Waste Authority Regional Waste Management Center at SRS accepts nonhazardous and nonradioactive solid waste from SRS and eight surrounding South Carolina counties. This municipal solid waste landfill provides state-of-the-art Subtitle D (nonhazardous) facilities for land-filling solid waste while reducing the environmental consequences associated with construction and operation of multiple county-level facilities. It was designed to accommodate combined SRS and county solid waste disposal needs for at least 20 years, with a projected maximum operational life of 45 to 60 years (DOE 2000). The landfill is designed to handle an average of 1,000 tons per day and a maximum of 2,000 tons per day of municipal solid waste. The SRS and eight cooperating counties had a combined generation rate of 900 tons per day in 1995. The Three Rivers Solid Waste Authority Regional Waste Management Center opened in mid-1998.

Activities supporting the treatment and management of sodium-bonded spent nuclear fuel and other planned SRS activities would not generate larger volumes of radioactive, hazardous, or solid waste beyond the current and projected capacities of SRS waste storage and/or management facilities.

4.11.3 Transportation

The Programmatic Spent Nuclear Fuel EIS (DOE 1995a) analyzed the cumulative impacts of all transportation of radioactive materials, including impacts from reasonably foreseeable future actions that include transportation of radioactive material for a specific purpose and general radioactive materials transportation that is not related to a particular action. The total worker and general population collective doses are expected to be less than 1 person-rem. The impacts of this program are quite small compared with overall transportation impacts. 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 latent cancer fatalities) for the period 1943 through 2035 (93 years).

4.12 PROGRAMMATIC CONSIDERATIONS

Programmatic considerations presented in this section provide information on the regulatory environment applicable to spent nuclear fuel and high-level radioactive waste. Also presented are schedule considerations for the disposal of DOE-owned spent nuclear fuel or high-level radioactive waste in a geologic repository.

4.12.1 Regulatory Environment Considerations

Prior to the acceptance of spent nuclear fuel or high-level radioactive waste at a geologic repository, certain NRC and EPA regulatory requirements and DOE Office of Civilian Radioactive Waste Management requirements must be met. Regulatory requirements specific to DOE's sodium-bonded spent nuclear fuel, are identified in the Civilian Radioactive Waste Management Office's current April 1999, Draft Waste Acceptance System Requirements Document (DOE 1999c).

One of the key NRC requirements for acceptance of spent nuclear fuel or high-level radioactive waste is that it cannot contain or generate materials that are explosive, pyrophoric, or chemically reactive (in the repository environment) in a form or amount that could compromise the repository's ability to perform its waste isolation function or to satisfy its performance objective [10 CFR 60.135(b)(1)]. The No Action Alternative may not satisfy this requirement, because the metallic sodium is highly reactive. The metallic uranium is also reactive and potentially pyrophoric, and in some cases the fuel contains highly enriched uranium, which would require criticality control measures. It also is uncertain whether the treatment technology, identified for the blanket spent nuclear fuel under Alternative 2 (cleaning the fuel to remove sodium and packaging in a high-integrity can), would be adequate to meet this NRC requirement. Under all other action alternatives, this requirement could be met.

The Waste Acceptance System Requirements Document identifies the Civilian Radioactive Waste Management Office's acceptance criteria for spent nuclear fuel and high-level radioactive waste. Under these criteria, the Civilian Radioactive Waste Management Office states that only spent nuclear fuel and high-level radioactive waste that is not subject to regulation under RCRA, Subtitle C, will be accepted for disposal. Untreated sodium-bonded spent nuclear fuel may be regulated under RCRA, Subtitle C, because it exhibits the characteristic of reactivity when exposed to water (40 CFR 261.23 (a)(2), (3)) and is ignitable (40 CFR 261.21 (a)(2)).

Under RCRA, 40 CFR 268.9 (c), "... no prohibited waste which exhibits a characteristic under 40 CFR part 261, subpart C, may be land disposed of unless the waste complies with the treatment standards under 40 CFR 268, subpart D." Deactivation is the waste treatment technology for waste that exhibits the characteristic of reactivity and ignitability (40 CFR 268.40). RCRA land disposal requirements (i.e., 40 CFR 268.40) also require generators of waste that exhibits the characteristics of reactivity to water or ignitability to identify all underlying hazardous constituents reasonably expected to be present in the waste at the point of generation, and to treat these constituents to the universal treatment standards. If the characteristic waste is treated by the

applicable treatment technology and the waste no longer exhibits the characteristic, then the waste no longer needs to be regulated under RCRA, Subtitle C, and can be managed as a nonhazardous waste (62 *FR* 62083).

The direct disposal option of the No Action Alternative may not satisfy this requirement, because the sodium-bonded spent nuclear fuel could be considered to be reactive and ignitable, and therefore, it may not be accepted for disposal at a geologic repository. All of the alternatives under the proposed action would be able to deactivate the sodium-bonded fuel and remove the characteristics of reactivity and ignitability. The metallic uranium is considered to be reactive, and in some cases pyrophoric; however, it would not be a RCRA hazardous waste because it is defined under the Atomic Energy Act of 1954, as amended (42 U.S.C. 2001 *et seq.*), as a source, special nuclear, or byproduct material and, therefore, is excluded from RCRA under 40 CFR 261.4 (a)(4).

The Waste Acceptance System Requirements Document also identifies specific acceptance criteria for DOE's spent nuclear fuel and high-level radioactive waste. For high-level radioactive waste, the Civilian Radioactive Waste Management Office specifies borosilicate glass as a standard vitrified high-level radioactive waste form. For DOE's spent nuclear fuel, specific acceptance criteria have been developed for canistered DOE spent nuclear fuel, including naval spent nuclear fuel that is intended for disposal in the canister. Performance criteria for the ceramic high-level radioactive waste and the metallic high-level radioactive waste forms are being developed. However, no specific acceptance criteria have been developed for spent nuclear fuel that has been melted into a liquid form and then solidified. The No Action Alternative may be able to meet this requirement for the disposal canisters; however, as previously discussed, it may not meet the other waste acceptance requirements (e.g., NRC and RCRA regulations).

For Alternative 3 (blanket fuel) where the treated waste form would be a vitrified borosilicate glass, the specific acceptance criteria have been developed. However, final approval of this waste form would be contingent upon the requirements in the disposal facility's license.

For Alternative 1 (blanket and driver fuel), Alternative 2 (driver fuel), Alternative 3 (driver fuel), Alternative 4 (driver fuel), and Alternative 5 (driver fuel), performance criteria for the ceramic high-level radioactive waste and the metallic high-level radioactive waste form have been developed, but need approval. Again, final approval of this waste form would be contingent upon the requirements in the disposal facility's license.

For Alternative 2 (blanket fuel), the specific acceptance criteria for canistered spent nuclear fuel would apply and may be achieved. However, the long-term durability of the proposed overpack container has not been demonstrated or documented. Without such demonstration of extended containment, the ability of the high-integrity can concept to meet the safety standards proposed by the National Research Council is unknown (National Research Council 1998).

For Alternative 4 (blanket fuel), Alternative 5 (blanket fuel), and Alternative 6 (blanket and driver fuel), the specific acceptance criteria for conditioned spent nuclear fuel would need to be developed and approved.

4.12.2 Schedule Considerations

The schedule perspective for each of the alternatives is affected by two time frames: the proposed schedule for the construction, operation, and closure of a geologic repository, and 2035, the year by which DOE committed to remove all spent nuclear fuel from Idaho under the 1995 agreement with the State of Idaho.

The proposed schedule for the repository is discussed in the *Viability Assessment of a Repository at Yucca Mountain* (DOE 1998f). A site recommendation decision for the geologic repository is scheduled for 2001. If the site were to be subsequently authorized, a license application could be submitted in 2002. The NRC construction authorization decision could occur in 2005 at the earliest. Repository construction would begin

upon receipt of this authorization. DOE must update its licensing application and submit it to the NRC before the Commission will issue a license to receive and process nuclear waste. This update is scheduled for 2008. Assuming repository construction sufficient to begin waste emplacement will take five years, the first waste emplacement at Yucca Mountain could occur in 2010. DOE would design the repository to close as early as approximately 10 years after emplacement of the last waste package, or to be kept open for at least 100 years after initiation of waste emplacement, with a reasonable expectation that the repository actually could be kept open with appropriate maintenance for 300 years after initiation of waste emplacement. The Viability Assessment (DOE 1998f) assumes a reference case in which closure of a monitored geologic repository is initiated in 2110, 100 years after initiation of waste emplacement operations.

Under the No Action Alternative, the untreated sodium-bonded spent nuclear fuel could remain in storage at the current locations until 2035. After that, it would need to be transported outside the State of Idaho and stored or treated at another DOE site. If the waste acceptance criteria are finalized by 2010 and indicate that direct disposal of the sodium-bonded spent nuclear fuel is possible, the fuel could be packaged for direct disposal well before 2035.

The treatment of the driver spent nuclear fuel using the electrometallurgical technology under Alternatives 1 through 5 could start as early as 2000 and could be completed by 2006 to 2007. If the decision to select a technology is delayed until after 2010, when waste acceptance criteria may be finalized, it would require two to three years lead time for the reactivation or installation of new equipment for the electrometallurgical treatment technology and six to seven years for the processing, for a total of approximately 10 years. The high-level radioactive waste would be ready for disposal by 2020.

The treatment of driver spent nuclear fuel only using the melt and dilute process at ANL-W could start as early as 2005 and could be completed by 2007. If installation of the necessary equipment is delayed until after 2010, the conditioned spent nuclear fuel would be ready for disposal in 2017.

The treatment of the blanket spent nuclear fuel using the electrometallurgical technology under Alternative 1 could start as early as 2000 and could be completed by 2012 or 2013. Delaying a decision until after 2010 would add 10 to 15 years, depending on the time required to reactivate or install new equipment. The process still could be completed by 2030.

The preparation of the blanket spent nuclear fuel and its placement in high-integrity cans under Alternative 2 could start in 2003. Cleaning and sodium removal activities and packaging would take approximately six years and could be completed by 2009. Delaying a decision until after 2010 would delay the completion of this effort to approximately 2020.

The treatment of blanket spent nuclear fuel using the PUREX process at SRS would not start until 2005 because the F-Canyon is committed to other missions. Once started, however, all blanket spent nuclear fuel could be processed in less than one year. The decladding and sodium removal activities at ANL-W to prepare the blanket spent nuclear fuel for transportation and processing also would not start until 2003, allowing for installation of new equipment. It is estimated that preparation activities at ANL-W for all blanket spent nuclear fuel would last approximately six years. Therefore, the overall process could be completed by approximately 2010. At this time it is not clear whether the decision to process blanket spent nuclear fuel at the F-Canyon could be delayed until after 2010 because DOE has scheduled operations for the F-Canyon until 2005. If there were a programmatic decision to close the F-Canyon after currently scheduled operations are completed, the F-Canyon would not be available.

The preparation of blanket spent nuclear fuel for the melt and dilute process at ANL-W under Alternative 4 could start in 2003, allowing time for the installation of new equipment. The melt and dilute activities could start in 2005 and could be completed in seven years (by 2012). The process would require sodium removal

activities at ANL-W, which could be done in parallel. The blanket spent nuclear fuel preparation activities would start in 2003 and would require approximately six years for completion. The overall process could be completed by 2012. If a decision were delayed until after 2010, treatment would not be completed until about 2020.

The treatment of blanket spent nuclear fuel using the melt and dilute process at SRS under Alternative 5 could start after 2020 if capacity becomes available. It is estimated that the facility would be operational by 2005, but it is committed to other missions until 2035, as stated in the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000). If additional capacity becomes available, treatment could start as soon as 2020. Processing of the blanket spent nuclear fuel at SRS would take approximately three years. The decladding and sodium removal activities at ANL-W that are needed to prepare the fuel could start in 2003 and could be completed by 2009, well before processing begins. Delaying a decision until 2010 would push the completion of the decladding activities to 2019, which would be well before processing could begin at SRS.

The treatment of blanket and driver spent nuclear fuel using the melt and dilute process at ANL-W under Alternative 6 could start as early as 2003 and could be completed by 2015. Delaying a decision until 2010 would push completion to approximately 2025.

Table 4–73 summarizes the dates for completing the process for each alternative, given that a decision to proceed is made in the year 2000 or the year 2010.

Table 4–73 Treatment Completion Year

Alternatives [a]	*Decision in 2000*	*Decision in 2010*
No Action (direct disposal)	Before 2035	Before 2035
Alternative 1 Driver (only) Driver and blanket	 2006 2012	 2020 2030
Alternative 2 Driver Blanket	 2006 2009	 2020 2020
Alternative 3 Driver Blanket	 2006 2010	 2020 F-Canyon may not be available
Alternative 4 Driver Blanket	 2006 2012	 2020 2020
Alternative 5 Driver Blanket	 2006 2025	 2020 2025
Alternative 6 Driver (only) Driver and blanket	 2007 2015	 2017 2025

[a] See Section 2.5 for an explanation of alternatives.

4.13 Mitigation Measures

Following completion of an EIS and its associated Record of Decision, DOE is required to prepare a Mitigation Action Plan to address any mitigation commitments expressed in the Record of Decision (10 CFR 1021.331). The purpose of the Mitigation Action Plan is to explain how measures designed to mitigate adverse

environmental impacts will be planned and implemented. The Mitigation Action Plan is prepared prior to DOE taking any action directed by the Record of Decision that is the subject of a mitigation commitment.

Based on analyses of the environmental consequences of the proposed action presented earlier in this chapter, no mitigation measures would be necessary since all potential environmental impacts would be small and well within applicable requirements. Each DOE site would follow installation and operational practices that would minimize any potential impacts to air and surface water quality, noise, operational and public health and safety, and accident prevention and mitigation. These practices are dictated by Federal and state licensing and permitting requirements, as described in Chapter 5.

4.14 RESOURCE COMMITMENTS

This section describes the unavoidable adverse environmental impacts that could result from the proposed action; the relationship between short-term uses of the environment and the maintenance and enhancement of long-term productivity; and irreversible and irretrievable commitments of resources. Unavoidable adverse environmental impacts are impacts that would occur after implementation of all feasible mitigation measures. The relationship between short-term uses of the environment and the maintenance and enhancement of long-term productivity addresses issues associated with the condition and maintenance of existing environmental resources used to support the proposed action and the utility of these resources after their use. Resources that would be irreversibly and irretrievably committed are those that cannot be recovered or recycled and those that are consumed or reduced to unrecoverable forms.

4.14.1 Unavoidable Adverse Environmental Impacts

Implementing any of the alternatives considered in this EIS for the treatment and management of sodium-bonded spent nuclear fuel would result in unavoidable adverse impacts to the human environment. In general, these impacts are expected to be minimal and would come from incremental impacts attributed to the operation of treatment and management facilities at ANL-W and SRS.

Operation of treatment and management facilities at ANL-W and SRS would result in unavoidable increases of radiation exposures to workers and the general public. Workers would be exposed to direct radiation and other chemicals associated with the handling and treatment of the sodium-bonded spent nuclear fuel. The incremental annual dose contribution from the treatment and management of sodium-bonded spent nuclear fuel to the maximally exposed offsite individual, general population, and workers are discussed in Sections 4.3.4, 4.4.4, 4.5.4, 4.6.4, 4.7.4, and 4.8.4.

Also unavoidable would be the generation of additional low-level transuranic and mixed radioactive waste compared to baseline generation rates, which would either be treated and stored on site at ANL-W or SRS, or transported and managed off site at appropriate waste disposal facilities. Any other waste generated during treatment and management activities would be collected at the site, treated and/or stored, and eventually removed for suitable recycling or disposal off site in accordance with applicable EPA regulations.

Operation of treatment and management facilities at ANL-W and SRS would have minimal unavoidable adverse environmental impacts to air and water quality. Air quality would be affected by increases in various chemical or radiological constituents in the routine emissions typical of facility operations at these sites. Impacts to water resources and quality also would be affected by the release of various chemical or radiological constituents in the routine effluent only from PUREX processing at SRS. Impacts to the environment associated with the normal operation of facilities at ANL-W and SRS would occur regardless of the treatment and management of spent nuclear fuel. These routine impacts also have been addressed in various other NEPA documentation at these sites.

The alternative treatment processes would generate varying amounts of waste material that could affect storage requirements. This would be an unavoidable impact on the amount of available and anticipated storage space and the requirements of disposal facilities.

4.14.2 Relationship Between Local Short-Term Uses of the Environment and the Maintenance and Enhancement of Long-Term Productivity

Implementation of the alternatives, including the No Action Alternative, would cause short-term commitments of resources (e.g., air emissions and water discharges) and would permanently commit certain resources (e.g., dilution materials and energy). For each alternative, the short-term use of these resources would result in potential long-term benefits to the environment and the enhancement of long-term productivity by decreasing overall health risks to workers, the public, and the surrounding environment by reducing their exposure to hazardous and radioactive substances. The short-term effect on workers, the public, and the environment from the treatment of sodium-bonded spent nuclear fuel would be offset by the long-term benefits of safe, stable, secure storage of these materials.

Under the No Action Alternative, environmental resources already have been committed to the storage of spent nuclear fuel. This commitment would serve to maintain existing environmental conditions with little or no impacts to the long-term productivity of the environment. The continued storage of sodium-bonded spent nuclear fuel at ANL-W and INEEL until 2035 and the potential for its direct disposal in a geologic repository would result in less exposure to hazardous and radioactive materials for workers, the public, and the environment than would be experienced under the proposed action. Only the direct disposal of the sodium-bonded fuel in a repository would have the potential to enhance the long-term viability of the environment in Idaho.

Under the proposed action, the short-term use of environmental resources at ANL-W and SRS would be greater than for the No Action Alternative. The short-term commitment of resources would include the space required for onsite processing, the commitment of processing facilities, transportation, and other disposal resources and materials for the treatment and management of sodium-bonded spent nuclear fuel. Workers, the public, and the environment would be exposed to larger amounts of hazardous and radioactive materials over the short-term from the handling and treatment of the spent nuclear fuel, including process emissions and the handling of waste. Again, these commitments would be offset by an even greater potential for enhanced long-term viability of the environment than under the No Action Alternative.

Over the life of the proposed action, groundwater would be used at SRS to meet sanitary and process needs. After use and treatment, this water would be discharged into surface water streams. Depending on the site chosen (F- or L-Area) and the technology implemented over the short-term, the resulting increases in pollutant loadings would take advantage of the natural assimilative capacity of the receiving stream(s). However, these incremental pollutant loadings should not adversely affect either short- or long-term viability of the aquatic ecosystem. These impacts would be assessed during the regulatory permitting process once an alternative has been selected.

Regardless of location, air emissions associated with implementation of any of the technologies would add small amounts of radiological and nonradiological constituents to the air of the regions around ANL-W and SRS. During the project's life, these emissions would result in additional loading and exposure, but would not impact compliance with air quality or radiation exposure standards at either site. There would be no significant residual environmental effects to long-term environmental viability.

The management and disposal of sanitary solid waste and nonrecyclable radiological waste over the project's life would require energy and space at ANL-W and SRS treatment, storage, or disposal facilities. The land required to meet the solid waste needs would require a long-term commitment of terrestrial resources. Upon

the facilities' closures, DOE could decontaminate and decommission the facilities and/or equipment and restore them to brown field sites which could be available for future commercial or industrial development.

Regardless of location, continued employment, expenditures, and tax revenues generated during the implementation of any of the alternatives would directly benefit the local, regional, and state economies over the short-term. Long-term economic productivity could be facilitated by local governments investing project-generated tax revenues into infrastructure and other required services.

The use of short-term resources to operate spent nuclear fuel treatment and management facilities at either ANL-W or SRS would not affect the long-term productivity of these sites.

4.14.3 Irreversible and Irretrievable Commitments of Resources

Irreversible and irretrievable commitments of resources for each alternative potentially would include mineral resources during the life of the project and energy used in treating the waste. The commitment of capital, energy, labor, and material during the implementation of the alternatives generally would be irreversible.

Energy expended would be in the form of fuel for equipment and vehicles, electricity for facility operations, and human labor. The energy consumption of treatment and management facilities would be a small fraction of the total energy used at each DOE site. None of the technologies evaluated in the EIS would require significantly higher or lower energy consumption. Assuming that these facilities are totally dedicated to the treatment and management of sodium-bonded spent nuclear fuel, it is estimated that total electrical energy consumption would range from 101,500 megawatt hours for Alternative 2, high-integrity cans, to 130,000 megawatt hours for Alternative 5, melt and dilute at SRS. Operation of any proposed facility would generate nonrecyclable waste streams, such as radiological and nonradiological solid waste and some process wastewaters. However, certain materials and equipment used during operation of the proposed facility could be recycled when the facility is decontaminated and decommissioned.

The implementation of the alternatives considered in this EIS, including the No Action Alternative, would require water, electricity, steam, and diesel fuel. Water at SRS and ANL-W would be obtained from onsite groundwater sources and steam from existing onsite sources. Electricity and diesel fuel would be purchased from commercial sources. These commodities are readily available and the amounts required would not have an appreciable impact on available supplies or capacities. From a materials and energy resource commitment perspective, electrometallurgical treatment and PUREX process technologies would recover low-enriched uranium, which is usable as commercial reactor fuel.

The disposal of hazardous and/or radioactive waste also would cause irreversible and irretrievable commitments of land, mineral, and energy resources. Hazardous waste and low-level radioactive waste disposal would irreversibly and irretrievably commit land for its disposal. For each of the alternatives analyzed in this document, the No Action Alternative would have the least commitment of land, mineral, and energy resources.

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Chapter 5

Environmental Laws, Regulations, and Consultations

5. ENVIRONMENTAL LAWS, REGULATIONS, AND CONSULTATIONS

Chapter 5 presents the laws, environmental regulations, and consultations that apply to the proposed action and alternatives. Federal, state, and U.S. Department of Energy environmental, safety, and health laws, regulations, and directives are summarized in Section 5.1. Radioactive material packaging and transportation regulations are discussed in Section 5.2. Emergency management and response laws, regulations, and Executive Orders are discussed in Section 5.3. Consultations with Federal, state, and local agencies and Federally recognized Native American groups are discussed in Section 5.4.

5.1 Environmental, Safety, and Health Laws, Regulations, Executive Orders, and DOE Orders

There are a number of Federal environmental laws dealing with environmental protection, compliance, or consultation that affect compliance at every U.S. Department of Energy (DOE) location. In addition, certain environmental requirements have been delegated to state authorities for enforcement and implementation. It is DOE policy to conduct its operations in a manner that ensures protection of public health, safety, and the environment through compliance with all applicable Federal and state laws, regulations, Orders, and other requirements. This chapter describes the environmental, safety, and health laws, regulations, and Executive and DOE Orders that are important to DOE's implementation of the proposed action. The applicability of these laws, regulations, and Orders and how they affect the proposed action are discussed in Chapters 1, 2, 3, and 4, and the appendices, where appropriate. Appendix B discusses regulations that pertain to the methodologies used in the environmental impact statement (EIS) analyses. Appendices E and F discuss applicable health and safety regulations. Appendix F discusses relevant transportation regulations.

5.1.1 Federal Laws and Regulations

National Environmental Policy Act of 1969, as amended (42 U.S.C. 4321 *et seq.*)—The National Environmental Policy Act (NEPA) establishes a national policy promoting awareness of the environmental consequences of human activity on the environment and consideration of environmental impacts during the planning and decision-making stages of a project. It requires Federal agencies to prepare a detailed EIS for any major Federal action that could have a significant environmental impact.

DOE has prepared this EIS in accordance with the requirements of NEPA or as implemented by Council on Environmental Quality regulations (40 CFR 1500 *et seq.*) and DOE NEPA implementing regulations (10 CFR 1021).

Atomic Energy Act of 1954 (42 U.S.C. 2011 *et seq.*)—The Atomic Energy Act authorizes DOE to establish standards to protect health or minimize dangers to life or property for activities under DOE's jurisdiction. Through a series of DOE Orders, an extensive system of standards and requirements has been established to ensure safe operation of facilities. DOE regulations generally are found in Title 10 of the Code of Federal Regulations (CFR).

The Nuclear Waste Policy Act of 1982 (U.S.C. 10101 through 10271)—The Nuclear Waste Policy Act established the Office of Civilian Radioactive Waste Management (42 U.S.C. 10224) and the Nuclear Waste Fund (42 U.S.C. 10222) and defined its mission to develop a Federal system for the management and geologic disposal of commercial spent nuclear fuel and other high-level radioactive wastes. As originally enacted, it

called for the Secretary of Energy to recommend candidate repository sites, but in 1987 it was amended to require DOE to proceed with characterization of the Yucca Mountain site only (42 U.S.C. 10133 and 10172). The Energy Policy Act of 1992, Section 801, directed the U.S. Environmental Protection Agency (EPA) to promulgate public health and safety standards for the protection of the public from releases from radioactive materials stored or disposed of in the proposed repository at the Yucca Mountain site.

Low-Level Radioactive Waste Policy Act of 1980, as amended (42 U.S.C. 2021 *et seq.*)—This Act amended the Atomic Energy Act to specify that the Federal Government is responsible for disposal of low-level radioactive waste generated by its activities, and the states are responsible for disposal of other low-level radioactive waste. It provides for and encourages interstate compacts to carry out the state responsibilities.

Solid Waste Disposal Act of 1965, as amended by the Resource Conservation and Recovery Act of 1976 and the Hazardous and Solid Waste Amendments of 1984 (42 U.S.C. 6901 *et seq.*)—The Solid Waste Disposal Act of 1965, as amended, governs the transportation, treatment, storage, and disposal of hazardous and nonhazardous waste. Under the Resource Conservation and Recovery Act of 1976 (RCRA), which amended the Solid Waste Disposal Act of 1965, the EPA defines and identifies hazardous waste; establishes standards for its transportation, treatment, storage, and disposal; and requires permits for persons engaged in hazardous waste activities. Section 3006 of the Act (42 U.S.C. 6926) allows states to establish and administer those permit programs with EPA approval. The EPA regulations implementing RCRA are found in 40 CFR Parts 260 through 283.

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.

Federal Facilities Compliance Act of 1992 (42 U.S.C. 6961 *et seq.*)—Section 102(a)(3) of the Federal Facilities Compliance Act waives sovereign immunity for Federal facilities for fines and penalties for RCRA violations and state, interstate, and local hazardous and solid waste management requirements. This waiver was delayed for three years following enactment for violations of the land disposal restrictions storage and prohibition (RCRA section 3004(j)) involving mixed waste at DOE facilities. The Act further delays the waiver of sovereign immunity beyond the three-year period at a facility if DOE is in compliance with an approved plan for developing treatment capacity and technologies for mixed waste generated or stored at the facility, as well as with an Order requiring compliance with the plan.

DOE and the State of Idaho have an approved plan, known as the "Site Treatment Plan," and associated consent order. Some of the waste being analyzed in this EIS has been designated for treatment according to terms in the Idaho National Engineering and Environmental Laboratory (INEEL) Site Treatment Plan. If DOE makes a decision based on this EIS that differs from that agreed to with the State of Idaho in the Site Treatment Plan, that Plan would be subject to renegotiation.

Toxic Substances Control Act of 1976 (15 U.S.C. 2601 *et seq.*)—The Toxic Substances Control Act provides the EPA with the authority to require testing of chemical substances entering the environment and to regulate them as necessary. The law complements and expands existing toxic substance laws, such as Section 112 of the Clean Air Act and Section 307 of the Clean Water Act. The Toxic Substances Control Act requires compliance with inventory reporting and chemical control provisions of the Act to protect the public from the risks of exposure to chemicals. The Act also imposes strict limitations on the use and disposal of polychlorinated biphenyls, chlorofluorocarbons, asbestos, dioxins, certain metal-working fluids, and hexavalent chromium. Some disposal activities under this Act might require a permit from EPA.

Clean Air Act of 1970 (42 U.S.C. 7401 *et seq.*)—The Clean Air Act 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 (42 U.S.C. 7418) requires that each Federal agency 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 Clean Air Act: (1) requires the EPA to establish National Ambient Air Quality Standards (NAAQS) as necessary to protect the public health, with an adequate margin of safety, from any known or anticipated adverse effects of a regulated pollutant (42 U.S.C. 7409 *et seq.*); (2) requires establishment of national standards of performance for new or modified stationary sources of atmospheric pollutants (42 U.S.C. 7411); (3) requires specific emission increases to be evaluated so as to prevent a significant deterioration in air quality (42 U.S.C. 7470 *et seq.*); and (4) requires specific standards for releases of hazardous air pollutants (including radionuclides) (42 U.S.C. 7412). These standards are implemented through state implementation plans developed by each state with EPA approval. The Clean Air Act requires sources to meet standards and obtain permits to satisfy these standards.

Hazardous air pollutants are substances that may cause health and environmental effects at low concentrations. Currently, 189 compounds have been identified as hazardous air pollutants. A major source is defined as any stationary source, or a group of stationary sources located within a contiguous area under common control, that emits or has the potential to emit at least 10 tons per year of any single hazardous air pollutant or 25 tons per year of a combination of pollutants.

The 1990 amendments to the Clean Air Act substantially revised the program to regulate potential emissions of hazardous air pollutants. The aim of the new control program is to require state-of-the-art pollution control technology on most existing and all new emission sources. These provisions regulate emissions by promulgating emissions limits reflecting use of the maximum achievable control technology. These emission limits are then incorporated into a facility's operating permit. Air emissions are regulated by the EPA under 40 CFR Parts 50 through 99.

Radionuclide emissions other than radon from DOE facilities are also covered under the National Emission Standards for Hazardous Air Pollutants program (40 CFR 61.90-97). To determine compliance with the standard, an effective dose equivalent value for the maximally exposed members of the public is calculated using EPA-approved sampling procedures, computer models, or other EPA-approved procedures. DOE is currently determining if a National Emission Standards for Hazardous Air Pollutants permit will be required for radiological emissions from any spent nuclear fuel treatment and management facilities at the Savannah River Site (SRS) (stacks, process vents, etc.).

Clean Water Act of 1972 (33 U.S.C. 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 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.

The Clean Water Act provides water quality standards for the Nation's waterways, guidelines and limitations for effluent discharges from point-source discharges, and the National Pollutant Discharge Elimination System (NPDES) permit program. The NPDES program is administered by the Water Management Division of the EPA pursuant to regulations in 40 CFR Part 122 *et seq.* Sections 401 through 405 of the Water Quality Act of 1987 added Section 402(p) to the Clean Water Act to require the EPA to establish regulations for permits for stormwater discharges associated with industrial activities. Stormwater provisions of the NPDES program are set forth at 40 CFR 122.26. Permit modifications are required if the discharge effluent is altered. DOE will apply for discharge permit for spent nuclear fuel treatment and management facilities at SRS if the treatment process results in discharges to waters of South Carolina.

Safe Drinking Water Act of 1974, as amended (42 U.S.C. 300(f) *et seq.*)—The primary objective of the Safe Drinking Water Act is to protect the quality of public drinking water supplies and sources of drinking water. The implementing regulations, administered by the EPA unless delegated to the states, establish standards applicable to public water systems. These regulations include maximum contaminant levels (including those for radioactivity) in public water systems, which are defined as water systems that have at least 15 service connections used by year-round residents or regularly serve at least 25 year-round residents. The EPA regulations implementing the Safe Drinking Water Act are found under 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 4 millirem per year beta activity (40 CFR 141.16(a)). 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.

The States of Idaho and South Carolina have received authorization from EPA to implement the public drinking water system program and the underground injection control program under the Safe Drinking Water Act. The Division of Environmental Quality, as a subdivision of the Idaho Department of Health and Welfare, sets forth monitoring and reporting requirements for inorganic and organic chemicals, and radiochemicals in Idaho. The South Carolina Department of Health and Environmental Control has established similar requirements for South Carolina.

The Safe Drinking Water Act also provides for designation of aquifers to be protected from degradation due to their importance as the sole source of drinking water. The Snake River Plain aquifer underlying INEEL has been designated as a sole source aquifer by EPA (40 FR 100-109, October 7, 1991) because groundwater supplies 100 percent of the drinking water consumed within the Eastern Snake River Plain and an alternative source or sources is not available.

Hazardous Material Transportation Act of 1975 (49 U.S.C. 5105 *et seq.*)—The Hazardous Material Transportation Act requires the U.S. Department of Transportation to prescribe uniform national regulations for transportation of hazardous materials (including radioactive materials). Most state and local regulations regarding such transportation that are not substantively the same as the Department of Transportation regulations are preempted (i.e., rendered void) (49 U.S.C. 5125). This, in effect, allows state and local governments only to enforce the Federal regulations, not to change or expand upon them.

This program is administered by the Research and Special Programs Administration of the Department of Transportation, which coordinates its regulations with those of the U.S. Nuclear Regulatory Commission (NRC), under the Atomic Energy Act, and with the EPA, under RCRA, when covering the same activities.

Individual states and Tribes often have their own statutes and/or regulations governing transportation of hazardous or radioactive materials. These laws might also be applicable to DOE transportation activities. An example of a local law that affects transportation of materials offsite from the INEEL is the Shoshone-Bannock Tribal Ordinance, the Nuclear Materials Transportation Act, ENVR 92-S5, which restricts transportation of radioactive materials across the Shoshone-Bannock Reservation.

National Historic Preservation Act of 1966, as amended (16 U.S.C. 470 *et seq.*)—The National Historic Preservation Act provides that sites with significant national historic value be placed on the *National Register of Historic Places,* which is maintained by the Secretary of the Interior. Section 110 of the Act requires Federal agencies to identify, evaluate, inventory, and protect National Register resources on properties under their control. No permits or certifications are required under the Act. However, if a particular Federal activity may impact a historic property resource, consultation with the Advisory Council on Historic Preservation is required under 16 U.S.C. 470(f). Such consultation usually generates a Memorandum of Agreement, including stipulations that must be followed to minimize adverse impacts.

Coordination with the state Historic Preservation Officer also is undertaken to ensure that potentially significant sites are identified properly and appropriate mitigative actions are implemented. DOE has notified respective State Historic Preservation Offices of its intent to consult on this project.

Endangered Species Act of 1973 (16 U.S.C. 1531 *et seq.*)—The Endangered Species Act is intended to prevent the further decline of endangered and threatened species and to restore these species and habitats. Section 7 of the Act requires Federal agencies that have reason to believe a prospective action may affect an endangered or threatened species or its habitat to consult with the U.S. Department of the Interior to ensure that the action does not jeopardize the species or destroy its habitat. If, despite reasonable and prudent measures to avoid or minimize such impacts, the species or its habitat would be jeopardized by the action, a review process is specified to determine whether the action may proceed. DOE has consulted with the U.S. Fish and Wildlife Service regarding impacts on any species listed under the Endangered Species Act.

American Indian Religious Freedom Act of 1978 (42 U.S.C. 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.

Occupational Safety and Health Act of 1970 (29 U.S.C. 651 *et seq.*)—The Occupational Safety and Health Act establishes standards for 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 (OSHA), a U.S. Department of Labor agency. Although OSHA and the EPA both have a mandate to reduce exposures to toxic substances, OSHA's jurisdiction is limited to safety and health conditions that exist in the workplace environment.

Under the Act, it is the duty of each employer to furnish employees a place of employment free of recognized hazards that are likely to cause death or serious physical harm. Employees have a duty to comply with the occupational safety and health standards and rules, regulations, and Orders issued under the Act. OSHA regulations (29 CFR) establish specific standards that tell employers what must be done to achieve a safe and healthful working environment. Government agencies, including DOE, are not technically subject to OSHA regulations, but are required under 29 U.S.C. 668 to establish their own occupational safety and health programs for their places of employment which are consistent with OSHA standards. 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 and 54831.A). DOE keeps and makes available the various records of minor illnesses, injuries, and work-related deaths as required by OSHA regulations.

Pollution Prevention Act of 1990 (42 U.S.C. 13101 *et seq.*)—The Pollution Prevention Act establishes a national policy for waste management and pollution control. Source reduction is given first preference, followed by environmentally safe recycling, with disposal or releases to the environment as a last resort. In response to the policies established by the Act, DOE committed to participation in the Superfund Amendments and Reauthorization Act, Section 313, EPA 33/50 Pollution Prevention Program. The goal for facilities involved in compliance with Section 313 was to achieve a 33 percent reduction (from a 1993 baseline) in the release of 17 priority chemicals by 1997. On August 3, 1993, President Clinton issued Executive Order 12856, which required DOE to achieve a 50 percent reduction in total releases of all toxic chemicals by December 31, 1999.

Noise Control Act of 1972, as amended (42 U.S.C. 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.

5.1.2 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. DOE has issued regulations (10 CFR 1021) and DOE Order 5440.1E for compliance with this Executive Order.

Executive Order 11593 (National Historic Preservation, May 13, 1971)—Executive Order 11593 directs Federal agencies to locate, inventory, and nominate properties under their jurisdiction or control to the *National Register of Historic Places* if those properties qualify. This process requires DOE to provide the Advisory Council on Historic Preservation the opportunity to comment on the possible impacts of the proposed activity on any potential eligible or listed resources.

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 government agencies to avoid any short- and long-term adverse impacts on wetlands wherever there is a practicable alternative. DOE requirements for compliance with flood plain and wetlands activity are codified in 10 CFR 1022.

Executive Order 12088 (Federal Compliance with Pollution Control Standards, October 13, 1978, as amended by Executive Order 12580, Federal Compliance with Pollution Control Standards, January 23, 1987)—Executive Order 12088 directs Federal agencies to comply with applicable administrative and procedural pollution control standards established by, but not limited to, the Clean Air Act, the Noise Control Act, the Clean Water Act, the Safe Drinking Water Act, the Toxic Substances Control Act, and RCRA.

Executive Order 12580 (Superfund Implementation)—Executive Order 12580 delegates to the heads of executive departments and agencies the responsibility for undertaking: (1) remedial actions for releases or threatened releases that are not on the National Priority List, and (2) removal actions, other than emergencies, where the release is from any facility under the jurisdiction or control of executive departments and agencies.

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 any disproportionately high and adverse human health or environmental effects of their programs, policies, and activities on minority or low-income populations.

Executive Order 13101 (Greening the Government Through Waste Prevention, Recycling, and Federal Acquisition)—Executive Order 13101 requires Federal agencies to incorporate waste prevention and recycling

in its daily operations and work to increase and expand markets for recovered materials. This Order states that it is national policy to prefer pollution prevention, whenever feasible. Pollution that cannot be prevented should be recycled; pollution that cannot be prevented or recycled should be treated in an environmentally safe manner. Disposal should be employed only as a last resort.

5.1.3 DOE Directives

The Atomic Energy Act authorizes DOE to establish standards to protect health or minimize dangers to life or property from activities under DOE's jurisdiction. Through a series of DOE Directives, an extensive system of policies, Orders, notices, manuals, and guides has been established to ensure safe operation of facilities.

DOE regulations are generally found in Title 10 of the Code of Federal Regulations. These regulations address such areas as energy conservation, administrative requirements and procedures, nuclear safety, and classified information. For the purpose of this EIS, relevant regulations include: Procedural Rules for DOE Nuclear Activities (10 CFR 820); Nuclear Safety Management (10 CFR 830); Radiation Protection of the Public and the Environment (10 CFR 834, Draft); Occupational Radiation Protection (10 CFR 835); Compliance with the National Environmental Policy Act (10 CFR 1021); and Compliance with Floodplains/Wetlands Environmental Review Requirements (10 CFR 1022).

DOE Directives are issued in support of health, safety, and environmental programs. Many of DOE's Directives are in the process of being revised and reorganized to reduce duplication and eliminate obsolete provisions. The new DOE Directives are organized by series, with each Directive identified by a letter and three digit number, and will include all DOE policies, Orders, notices, manuals, and guides. The remaining DOE Directives are expected to be revised and converted to the new DOE numbering system over the next two years. The major DOE Directives pertaining to the proposed action and alternatives are listed in **Table 5–1**.

5.1.4 State Environmental Laws, Regulations, and Agreements

Certain environmental requirements have been delegated to state authorities for implementation and enforcement. It is DOE policy to conduct its operations in an environmentally safe manner in compliance with all applicable laws, regulations, and standards, including state laws and regulations. A list of potentially applicable state laws, regulations, and agreements are provided in **Table 5–2**.

5.2 Radioactive Material Packaging and Transportation Regulations

Transportation of hazardous and radioactive materials and substances are governed by the Department of Transportation and the NRC. Department of Transportation regulations, which may be found under 49 CFR Parts 171 through 178 and 49 CFR Parts 383 through 397, contain requirements for identifying a material as hazardous or radioactive. These regulations interface with NRC 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.

The NRC regulations applicable to radioactive materials transportation may be found under 10 CFR Part 71. These regulations include 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. DOE may certify its own packages, per 49 CFR 173.7(d).

Table 5–1 Relevant DOE Directives

DOE Directive	Subject
Leadership/Management Planning	
O 151.1	Comprehensive Emergency Management System (09/25/95; Change 2, 08/21/96)
Information and Analysis	
O 231.1	Environment, Safety, and Health Reporting (09/30/95; Change 2, 11/07/96)
O 232.1A	Occurrence Reporting and Processing of Operations Information (07/21/97)
Work Processes	
O 414.1A	Quality Assurance (9/29/99)
O 420.1	Facility Safety (10/13/95; Change 2, 10/24/96)
O 435.1	Radioactive Waste Management (07/09/99)
O 440.1A	Worker Protection Management for DOE Federal and Contractor Employees (03/27/98)
N 441.1	Radiological Protection for DOE Activities (09/30/95, extended until 06/30/00 by N 441.4, 11/20/98)
N 441.4	Extension of DOE N 441.1, Radiological Protection for DOE Activities (11/20/98)
O 451.1A	National Environmental Policy Act Compliance Program (06/05/97)
O 460.1A	Packaging and Transportation Safety (10/02/96)
O 460.2	Departmental Materials Transportation and Packaging Management (09/27/95; Change 1, 10/26/95)
O 470.1	Safeguards and Security Program (09/28/95; Change 1, 06/21/96)
O 470.2	Safeguards and Security Independent Oversight Program (12/23/98)
O 474.1	Control and Accountability of Nuclear Materials (8/11/99)
Personnel Relations and Services	
3790.1B	Federal Employee Occupational Safety and Health Program (01/07/93)
Real Property Management	
4330.4B	Maintenance Management Program (02/10/94)
Project Management	
4700.1	Project Management System (03/06/87; Change 1, 06/02/92)
Environmental Quality and Impact	
5400.1	General Environmental Protection Program (11/09/88; Change 1, 06/29/90)
5400.5	Radiation Protection of the Public and the Environment (02/08/90; Change 2, 01/07/93)
5480.4	Environmental Protection, Safety, and Health Protection Standards (05/15/84; Change 4, 01/07/93)
5480.19	Conduct of Operations Requirements for DOE Facilities (07/09/90; Change 1, 05/18/92)
5480.20A	Personnel Selection, Qualification, and Training Requirements for DOE Nuclear Facilities (11/15/94)
5480.21	Unreviewed Safety Questions (12/24/91)
5480.22	Technical Safety Requirements (02/25/92; Change 2, 01/23/96)
5480.23	Nuclear Safety Analysis Report (04/10/92; Change 1, 03/10/94)
5480.30	Nuclear Reactor Safety Design Criteria (01/19/93)
5484.1	Environmental Protection, Safety, and Health Protection Information Reporting Requirements (02/24/81; Change 7, 10/17/90)
Emergency Preparedness	
5530.3	Radiological Assistance Program (01/14/92; Change 1, 04/10/92)
5530.5	Federal Radiological Monitoring and Assessment Center (07/10/92; Change 1 12/02/92)
Defense Programs	
5610.14	Transportation Safeguards System Program Operations (05/12/93)
5632.1C	Protection and Control of Safeguards and Security Interests (07/15/94)
5632.7A	Protective Force Program (04/13/94; Change 1, 02/13/95)
5660.1B	Management of Nuclear Materials (05/26/94)
Design	
6430.1A	General Design Criteria (04/06/89)

Table 5–2 State Environmental Laws, Regulations, and Agreements

Law/Regulation/Agreement	*Citation*	*Potential Requirements*
Idaho National Engineering Environmental Laboratory (INEEL), Idaho		
Idaho Environmental Protection and Health Act	ID Code, Title 39, Chapter 1	Provides for development of air pollution control permitting regulations.
Idaho Air Pollution Control Act	ID Code, Title 39, Chapter 29	Requires permitting of sources and control of toxic air pollutants and other pollutants.
Rules for the Control of Air Pollution in Idaho	IDAPA 16, Title 01, Chapter 01	Enforces national ambient air quality standards.
Idaho Water Pollution Control Act	ID Code, Title 39, Chapter 36	Enhances and preserves the quality and the value of water resources.
Idaho Rules for Public Drinking Water Systems	IDAPA 16, Title 01, Chapter 08	Controls and regulates the design, construction, operation, maintenance, and quality control of public drinking water.
Water Quality Standards and Wastewater Treatment Regulations	IDAPA 16, Title 01, Chapter 02	Enforces standards relating to the discharge of effluent into the water.
Transportation of Hazardous Waste	ID Code, Title 18, Chapter 39 ID Code, Title 49, Chapter 22	Regulates transportation of hazardous materials/hazardous waste on highways.
Various Acts Regarding Fish and Game	ID Code, Title 36, Chapters 9, 16 and 19	Requires consultation with responsible agency.
Endangered Species Act	ID Code, Title 67, Chapter 8	Requires consultation with Department of Fish and Game.
Classification and Protection of Wildlife	IDAPA 13, Title 01, Chapter 06	Requires consultation with Department of Fish and Game.
Idaho Historic Preservation	ID Code, Title 67, Chapters 41 and 46	Requires consultation with responsible local governing body.
Memorandum of Agreement	January 26, 1994	Requires consultation with Shoshone-Bannock Tribes.
Agreement-in-Principal (formerly Tribal Working Agreement)	August 6, 1998	Establishes understanding and commitment between the Tribes and DOE.
Federal Facility Agreement and Consent Order	December 9, 1991	Establishes a process for evaluating past potential releases to the environment at Idaho National Engineering and Environmental Laboratory (INEEL).
Spent Fuel Settlement Agreement (also known as the Batt Agreement)	October 16, 1995	Allows INEEL to receive spent nuclear fuel and mixed waste from off site and establishes schedules for the treatment of high-level radioactive waste, removal of spent nuclear fuel from the state, and treatment of mixed waste.

Law/Regulation/Agreement	Citation	Potential Requirements
Savannah River Site, South Carolina		
South Carolina Pollution Control Act	SC Code, Title 48, Chapter 1	Provides for the development of air pollution permitting regulations and air pollution control regulations
South Carolina Air Pollution Control Regulations and Standards	R.61-62	Requires permit prior to construction or modification of an air contaminant source and control of toxic air pollutants and other pollutants.
South Carolina Atomic Energy & Radiation Control Act	SC Code, Title 13, Chapter 7	Establishes standards for radioactive air emissions.
South Carolina Atomic Energy & Radiation Regulations and Standards	R.61-63 R.61-83	Establishes standards for radioactive air emissions.
South Carolina Pollution Control Act-Water	SC Code, Title 48, Chapter 1	Requires permit prior to construction or modification of a water discharge source.
South Carolina Water Pollution Control Regulations and Standards	R.61-9	Requires permit for the discharge of pollutants from any point source into waters of the state.
South Carolina Water Classification and Standards	R.61-68	Establishes official classified water uses, rules, and specific numeric water quality standards for protecting classified and existing water uses.
South Carolina Safe Drinking Water Act	SC Code, Title 44, Chapter 55	Establishes drinking water standards.
South Carolina Hazardous Waste Regulations and Standards	R.61-79 R.61-99 R.61-104	Protects human health and the environment by requiring careful management practices of hazardous waste.
South Carolina Solid Waste and Policy Management Act	SC Code, Title 44, Chapter 96	Establishes standards to treat, store, or dispose of solid waste.
South Carolina Solid Waste Regulations and Standards	R.61-107	Requires permit to store, collect, dispose, or transport solid waste.
South Carolina Nongame and Endangered Species Conservation Act	SC Code, Title 50, Chapter 15	Requires consultation with Wildlife and Marine Resources Department and minimization of impact.
South Carolina Museum Commission and Archaeology and Anthropology	Title 60, Chapter 12	Requires consultation with state Historic Preservation Office and minimization of impact.

Transportation casks, which are used to transport the radioactive material, are subject to numerous inspections and tests (10 CFR 71.87). These tests are designed to ensure that the cask components are properly assembled and meet applicable safety requirements. Tests and inspections are identified clearly in the Safety Analysis Report for Packaging and/or the Certificate of Compliance for each cask. Casks are loaded and inspected by registered users in compliance with approved quality assurance programs. Operations involving the casks are conducted in compliance with 10 CFR 71.91. Reports of defects or accidental mishandling are submitted to the NRC.

5.3 Emergency Management and Response Laws, Regulations, and Executive Orders

This section discusses the laws, regulations, and Executive Orders applicable to emergency management and response for the proposed action and alternatives.

5.3.1 Federal Laws

Emergency Planning and Community Right-to-Know Act of 1986 (U.S.C. 11001 *et seq.*) (also known as "SARA Title III")—This Act requires emergency planning and notice to communities and government agencies of the presence and release of specific chemicals. The EPA implements this Act under regulations found at 40 CFR Parts 355, 370, and 372. Under Subtitle A of this Act, Federal facilities are required to 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 emission reporting began in 1988. DOE requires compliance with Title III as a matter of DOE policy.

Comprehensive Environmental Response, Compensation, and Liability Act of 1980 (42 U.S.C. 9604(I) (also know as "Superfund")—This Act provides authority for Federal and state governments to respond directly to hazardous substances incidents. The Act requires reporting of spills, including radioactive spills, to the National Response Center.

Robert T. Stafford Disaster Relief and Emergency Assistance Act of 1988 (42 U.S.C. 5121)—This Act, 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" to provide Federal assistance under this Act. The President, in Executive Order 12148, delegated all functions except those in Section 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.

Justice Assistance Act of 1984 (42 U.S.C. 3701-3799)—This Act establishes Emergency Federal Law Enforcement Assistance to assist state and local governments in responding to a law enforcement emergency. The Act defines the term "law enforcement emergency" as an uncommon situation that requires law enforcement and is or threatens to become serious or of epidemic proportions, 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 (e.g., the Mount Saint Helens volcanic eruption) are eligible for Federal law enforcement assistance that includes funds, equipment, training, intelligence information, and personnel.

5.3.2 Federal Regulations

Quantities of Radioactive Materials Requiring Consideration of the Need for an Emergency Plan for Responding to a Release (10 CFR 30.72, Schedule C)—This section of the regulations provides a list that is the basis used by both the public and private sector to determine if the radiological materials they handle must have an emergency response plan for unscheduled releases, and is one of the threshold criteria documents for DOE Hazards Assessments required by DOE Order 5500.3A, *Planning and Preparedness for Operational Emergencies*. The Federal Radiological Emergency Response Plan, dated November 1995, primarily discusses offsite Federal response in support of state and local governments with jurisdiction during a peacetime radiological emergency.

Occupational Safety and Health Administration Emergency Response, Hazardous Waste Operations, and Worker Right to Know (29 CFR)—This regulation establishes the OSHA 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 1920.120), and hazards

communication (Section 1910.1200) that enable 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 and Communications, Emergency Response Information Requirements (49 CFR 172)—This regulation defines the regulatory requirements for marking, labeling, placarding, and documenting hazardous materials shipments. The regulation also specifies the requirements for providing hazardous material information and training.

5.3.3 Executive Orders

Executive Order 12148 (Federal Emergency Management, July 20, 1979)—Executive Order 12148 transfers functions and responsibilities associated with Federal emergency management to the Director of the Federal Emergency Management Agency. The Order assigns the Director 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 12656 (Assignment of Emergency Preparedness Responsibilities, November 1988)—Executive Order 12656 assigns emergency preparedness responsibilities to Federal departments and agencies.

5.4 CONSULTATIONS

Certain laws, such as the Endangered Species Act, the National Historic Preservation Act, and the American Indian Religious Freedom Act, recommended that consultation and coordination with other Federal agencies, state and local agencies, and Federally recognized Native American groups take place prior to a prospective action to ensure that the action does not jeopardize or destroy important resources. These consultations must occur on a timely basis before any proposed action can begin.

Consultations associated with the proposed action involve biotic resources, cultural resources, and Native American religious rights. Biotic resources consultations are to address the potential for the proposed action to disturb sensitive species or habitats. Cultural resources consultations are to address the potential disruption of important cultural resources and archaeological sites. Native American consultations are to address any potential disturbance of ancestral Native American sacred sites and traditional resources and practices. DOE consulted with the appropriate agencies, as discussed in Chapters 3 and 4.

Chapter 6

Glossary

6. GLOSSARY

Air Pollutant — Any substance in the air which could, if in a high-enough concentration, harm man, animals, vegetation, or material.

Air Quality Control Region — A geographic subdivision of the United States, designed to deal with pollution on a regional or local level. Some regions span more than one state.

Alluvial Deposits — Deposits of earth, sand, gravel, and other materials carried by moving surface water and deposited at points of weak water flow.

Alpha Particle — A positively charged particle, consisting of two protons and two neutrons, that is emitted during radioactive decay from the nucleus of certain nuclides. It is the least penetrating of the three common types of radiation (alpha, beta, and gamma).

Alpha Waste — Waste containing radioactive isotopes that decay by producing alpha particles.

Ambient Air — The surrounding atmosphere as it exists around people, plants, and structures. Air quality standards are used to provide a measure of the health-related and visual characteristics of the air.

Ambient Air Quality Standards — The level of pollutants in the air prescribed by regulations that may not be exceeded during a specified time in a defined area.

Aquatic — Living or growing in, on, or near water.

Aquifer — A saturated geologic unit through which significant quantities of water can migrate under natural hydraulic gradients.

Archaeological Resources — Any location where humans have altered the terrain or discarded artifacts during either prehistoric or historic times.

Artifact — An object produced or shaped by human workmanship of archaeological or historical interest.

As Low as Reasonably Achievable — A concept applied to ensure the quantity of radioactivity released to the environment and the radiation exposure of onsite workers in routine operations, including "anticipated operational occurrences," is maintained as low as reasonably achievable. It takes into account the state of technology, economics of improvements in relation to benefits to public health and safety, and other societal and economic considerations in relation to the use of nuclear energy in the public interest.

Background Radiation — Ionizing radiation present in the environment from cosmic rays and natural sources in the Earth; background radiation varies considerably with location. The U.S. average background radiation is 300 millirem per year.

Badged Worker — A worker who has the potential to be exposed to radiation and is equipped with a dosimeter to measure his/her dose.

Barrier — Any material or structure that prevents or substantially delays movement of radionuclides toward the accessible environment.

Baseline — A quantitative expression of conditions, costs, schedule, or technical progress to serve as a base or standard for measurement during the performance of an effort; the established plan against which the status of resources and progress of a project can be measured. For this environmental impact statement, the environmental baseline is the site environmental conditions as they exist or have been estimated to exist in the absence of the proposed action.

BEIR V — Biological Effects of Ionizing Radiation; referring to the fifth in a series of committee reports from the National Research Council.

Beta Particle — A charged particle emitted from the nucleus of an atom during radioactive decay. A negatively charged beta particle is identical to an electron; a positively charged beta particle is called a "positron."

Biota (biotic) — The plant and animal life of a region (pertaining to biota).

Blanket Fuel — Those fuel tubes or elements composed of depleted or natural enrichment of uranium, placed at the perimeter of the reactor core, and used to breed the fissile material plutonium-239 or used as shielding.

Blending — Selecting spent nuclear fuel assemblies of different characteristics for inclusion in a transportation cask, storage mode, or waste package; also, selecting high-level radioactive waste of different characteristics for inclusion in a transportation cask or waste package to meet design goals.

Borosilicate Waste Glass — A glass that typically contains approximately 20 to 40 weight percent waste oxides, 40 to 65 weight percent silica, 5 to 10 weight percent boron oxide, and 10 to 20 weight percent alkali oxides, plus other oxide constituents.

Breeder Reactor — A type of nuclear reactor that creates more fissionable fuel than it uses.

Burnup — A term used to indicate the amount of fuel consumed during the irradiation process. The percentage of heavy metal atoms fissioned or the thermal energy produced per mass of fuel (usually measured in megawatt days per ton [MWd/t]).

Calcine — To heat to a high temperature without fusing in order to decompose or oxidize; the material produced by converting high-level radioactive waste to unconsolidated granules or powder.

Cancer — The name given to a group of diseases characterized by uncontrolled cellular growth with cells having invasive characteristics such that the disease can transfer from one organ to another.

Canister — The structure surrounding the waste form (e.g., high-level radioactive waste immobilized in borosilicate glass) that facilitates handling, storage, transportation, and/or disposal. A canister is a metal receptacle with the following purpose: (1) for solidified high-level radioactive waste, it is used as a pour mold and (2) for spent nuclear fuel, it may provide structural support for intact spent nuclear fuel, loose rods, nonfuel components, or confinement of radionuclides.

Canning — The process of placing spent nuclear fuel in canisters to retard corrosion, contain radioactive releases, or control geometry.

Capable Fault — A fault that has exhibited one or more of the following characteristics:

(1) Movement at or near the ground surface at least once within the past 35,000 years or movement of a recurring nature within the past 500,000 years.

(2) Macro-seismicity instrumentally determined with records of sufficient precision to demonstrate a direct relationship with the fault.

(3) A structural relationship to a capable fault according to characteristics (1) or (2) of this paragraph such that movement on one could be reasonably expected to be accompanied by movement on the other.

Cask — A heavily shielded container that meets U.S. Nuclear Regulatory Commission and U.S. Department of Transportation regulatory requirements and is used to store and/or ship radioactive materials (i.e., spent nuclear fuel or high-level radioactive waste). Lead, depleted uranium, and steel are common materials used in the manufacture of casks.

Characterization — The determination of waste composition and properties, whether by review of process knowledge, nondestructive examination or assay, or sampling and analysis, generally done for the purpose of determining appropriate storage, treatment, handling, transport, and disposal requirements.

Chronic Exposure — Low-level radiation exposure incurred over a long time period due to residual contamination.

Cladding — The outer jacket of a fuel element, cladding is usually made of aluminum, stainless steel, or zirconium alloy. It is used to prevent fuel corrosion and retain fission products during reactor operation or to prevent releases into the environment during storage.

Class I Areas — National parks and wilderness areas designated by the Prevention of Significant Deterioration section of the Clean Air Act amendments. These amendments and the implementing regulations provide special protection to air quality and air quality-related values in such areas. Only very slight deterioration of air quality is allowed in Class I areas.

Class II Areas — Most of the country not designated as Class I is designated as Class II. Class II areas are generally cleaner than air quality standards require and moderate increases in new pollution are allowed after a regulatory-mandated impacts review.

Code of Federal Regulations (CFR) — All Federal regulations in force are published in codified form in the Code of Federal Regulations.

Collective Committed Effective Dose Equivalent — The committed effective dose equivalent of radiation for a population.

Committed Dose Equivalent — The predicted total dose equivalent to a tissue or organ over a 50-year period after an intake of a radionuclide into the body. It does not include external dose contributions. Committed dose equivalent is expressed in units of rem or sievert. The committed effective dose equivalent is the sum of the committed dose equivalents to various tissues of the body, each multiplied by the appropriate weighting factor.

Community (biotic) — All plants and animals occupying a specific area under relatively similar conditions.

Conditioning — Any process which prepares or treats spent nuclear fuel or high-level radioactive waste for storage, transportation, or disposal in accordance with regulatory requirements (e.g., melt and dilute product).

Conformity — Conformity is defined in the Clean Air Act as the action's compliance with an implementation plan's purpose of eliminating or reducing the severity and number of violations of the National Ambient Air Quality Standards and achieving expeditious attainment of such standards; and that such activities will not: (1) cause or contribute to any new violation of any standard in any area; (2) increase the frequency or severity of any existing violation of any standard in any area; or (3) delay timely attainment of any standard or any required interim emission reduction or other milestones in any area.

Consumptive Water Use — The difference in the volume of water withdrawn from a body of water and the amount released back into the body of water.

Colluvial Deposits — Loose deposits of rock and other material, usually at the foot of a slope or cliff and brought there chiefly by gravity.

Contact-handled Waste — Packaged waste whose external surface dose rates does not exceed 200 millirem per hour.

Container — With regard to radioactive waste, the metal envelope in the waste package that provides the primary containment function of the waste package and is designed to meet the containment requirements of 10 CFR 60.

Contamination — The deposition of unwanted radioactive material on the surfaces of structures, areas, objects, or personnel.

Coolant — A gas or liquid circulated through a nuclear reactor to remove or transfer heat.

Credible Accident — An accident that has a probability of occurrence greater than or equal to one in a million years.

Criteria Pollutants — The Clean Air Act required the U.S. Environmental Protection Agency to set air quality standards for common and widespread pollutants after preparing "criteria documents" summarizing scientific knowledge on their health effects. Today there are standards in effect for six "criteria pollutants": sulfur dioxide (SO_2), carbon monoxide (CO), particulate matter less than or equal to 10 microns in diameter (PM_{10}) and less than or equal to 2.5 microns in diameter ($PM_{2.5}$), nitrogen dioxide (NO_2), ozone (O_3), and lead (Pb).

Critical Habitat — Defined in the *Endangered Species Act* of 1973 as "specific areas within the geographical area occupied by [an endangered or threatened] species, essential to the conservation of the species and which may require special management considerations or protection; and specific areas outside the geographical area occupied by the species that are essential for the conservation of the species."

Criticality — A self-sustained nuclear chain reaction resulting from fissionable material of sufficient mass in a particular geometry.

Cultural Resources — Archaeological sites, historical sites, architectural features, traditional use areas, and Native American sacred sites.

Cumulative Impacts — In an environmental impact statement, 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 (Federal or nonfederal), private industry, or individual(s) undertakes such other actions. Cumulative impacts can result from individually minor but collectively significant actions taking place over a period of time (40 CFR 1508.7).

Curie (Ci) — A unit of radioactivity equal to 37 billion disintegrations per second; also a quantity of any nuclide or mixture of nuclides having 1 curie radioactivity.

Day-Night Average Sound Level — The 24-hour A-weighted (see decibel, A-weighted) equivalent sound level expressed in decibels, with a 10-decibel penalty added to sound levels between 10:00 p.m. and 7:00 a.m. to account for increased annoyance due to noise during nighttime hours.

Decay Heat (radioactivity) — The heat produced by the decay of certain radionuclides.

Decay (radioactive) — The decrease in the amount of any radioactive material with the passage of time due to the spontaneous transformation of an unstable nuclide into a different nuclide or into a different energy state of the same nuclide; the emission of nuclear radiation (alpha, beta, or gamma radiation) is part of the process.

Decladding — The process of mechanically removing the cladding from the fuel pin in a fuel element.

Decibel (dB) — A logarithmic unit of sound measurement which describes the magnitude of a particular quantity of sound pressure power with respect to a standard reference value. In general, a sound doubles in loudness for every increase of 10 decibels.

Decibel, A-weighted (dBA) — A unit of frequency weighted sound pressure level, measured by the use of a metering characteristic and the "A" weighting specified by the American National Standards Institution ANSI S1.4-1983 (R1594), that accounts for the frequency response of the human ear.

Deciduous — Trees which shed leaves at a certain season.

Decommissioning — The process of removing a facility from operation, followed by decontamination, entombment, dismantlement, or conversion to another use.

Decontamination — The actions taken to reduce or remove substances that pose a substantial present or potential hazard to human health or the environment, such as radioactive or chemical contamination from facilities, equipment, or soils by washing, heating, chemical or electrochemical action, mechanical cleaning, or other techniques.

Degraded (spent nuclear fuel) — Spent nuclear fuel whose external cladding has cracked, pitted, corroded, or potentially allows the leakage of radioactive materials.

°C (degrees Celsius) — A unit for measuring temperature using the centigrade scale in which the freezing point of water is 0 degrees and the boiling point is 100 degrees.

°F (degrees Fahrenheit) — A unit for measuring temperature using the Fahrenheit scale in which the freezing point of water is 32 degrees and the boiling point is 212 degrees.

Depleted Uranium — Uranium with a smaller percentage of uranium-235 than the 0.711 weight percent found in natural uranium. It is a byproduct of the uranium enrichment process, during which uranium-235 is collected from one batch of uranium, thereby depleting it, and adding to another batch to increase its concentration of uranium-235.

Dilute — To reduce the concentration of a substance by adding another material to it.

Disposal — The isolation of radioactive waste from the accessible environment, as defined in 10 CFR 60.2. Disposal means the emplacement in a repository of high-level radioactive waste, spent nuclear fuel, or other highly radioactive material with no foreseeable intent of recovery, whether or not such emplacement permits the recovery of such waste.

Direct Jobs — The number of workers required at a site to implement an alternative.

Disassembly — Removal of the fuel elements from the fuel assembly.

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.

DOE Site Boundary — A geographic boundary within which public access is controlled and activities are governed by the U.S. Department of Energy (DOE) and its contractors, not by local authorities. Based on the definition of exclusion zone, a public road traversing a DOE site is considered to be within the DOE site boundary if DOE or the site contractor has the capability to control the road at any time necessary.

Dose — The energy imparted to matter by ionizing radiation. The unit of absorbed dose is the rad.

Dose Commitment — The dose an organ or tissue would receive during a specified period of time (e.g., 50 to 100 years) as a result of intake (by ingestion or inhalation) of one or more radionuclides from a defined release, frequently over a year's time.

Dose Equivalent — The product of absorbed dose in rad (or Gray) and a quality factor, which quantifies the effect of this type of radiation in tissue. Dose equivalent is expressed in units of rem or Sievert, where 1 rem equals 0.01 Sievert.

Dosimeter — A small device (instrument) carried by a radiation worker that measures cumulative radiation dose (e.g., film badge or ionization chamber).

Drinking Water Standards — The level of constituents or characteristics in a drinking water supply specified in regulations under the Safe Drinking Water Act as the maximum permissible.

Driver Fuel — These fuel tubes or assemblies usually contain enriched uranium, plutonium, or thorium materials, which can be fissioned (or split) by neutrons. Because this fuel drives neutron bombardment of targets or blanket in a production, breeder, or research reactor, it is called driver fuel.

Dry Storage — Storage of spent nuclear fuel in environments where the fuel is not immersed in liquid for purposes of cooling and/or shielding.

Effective Dose Equivalent — The sum of the products of the dose equivalent received by specified tissues of the body and a tissue-specific weighting factor. This sum is a risk-equivalent value and can be used to estimate the health effects risk to the exposed individual. The tissue-specific weighting factor represents the fraction of the total health risk resulting from uniform whole-body irradiation that would be contributed by that particular tissue. The effective dose equivalent includes the committed effective dose equivalent from internal deposition of radionuclides, and the effective dose equivalent due to penetrating radiation from sources external to the body. Effective dose equivalent is expressed in units of rem or Sievert.

Effluent — Gas or fluid discharged into the environment.

Effluent (liquid) — Wastewater, treated or untreated, that flows out of a treatment plant, sewer, or industrial outfall; generally refers to waste discharged into surface waters.

Electrometallurgical Treatment — A technique to collect, concentrate, and immobilize fission products and transuranic elements from metallic spent nuclear fuel by removing the uranium in the spent fuel with an electrochemical cell. The treatment alters the chemical and physical nature of spent nuclear fuel to reduce its toxicity, volume, and mobility to render it amendable to transport, storage, or disposal.

Emergency Condition — For a nuclear facility, occurrences or accidents that might occur infrequently during startup testing or operation of the facility. Equipment, components, and structures might be deformed by these conditions to the extent that repair is required prior to reuse.

Emission — A material discharged into the atmosphere from a source operation or activity.

Emission Standards — Legally enforceable limits on the quantities and/or kinds of air contaminants that may be emitted into the atmosphere.

Empirical — Something that is based on actual measurement, observation, or experience rather than on theory.

Endangered Species — Any species which is in danger of extinction throughout all or significant portions of its range. The Endangered Species Act of 1973, as amended, establishes procedures for placing species on the Federal lists of endangered or threatened species.

Enriched Uranium — Uranium in which the abundance of the isotope uranium-235 is increased above the normal (naturally occurring) level of 0.711 weight percent.

Entrainment — The involuntary capture and inclusion of organisms in streams of flowing water; a term often applied to the cooling water systems of power plants/reactors. The organisms involved may include phyto- and zooplankton, fish eggs and larvae (ichthyoplankton), shellfish larvae, and other forms of aquatic life.

Environment, Safety, and Health Program — In the context of the U.S. Department of Energy (DOE), encompasses those DOE requirements, activities, and functions in the conduct of all DOE and DOE-controlled operations that are concerned with: impacts to the biosphere; compliance with environmental laws, regulations, and standards controlling air, water, and soil pollution; limiting the risks to the well-being of both the operating personnel and the general public; and protecting property against accidental loss or damage. Typical activities and functions related to this program include, but are not limited to, environmental protection, occupational safety, fire protection, industrial hygiene, health physics, occupational medicine, process and facilities safety, nuclear safety, emergency preparedness, quality assurance, and radioactive and hazardous waste management.

Environmental Assessment — A written environmental analysis prepared pursuant to the National Environmental Policy Act. This assessment is performed to determine whether a Federal action could significantly affect the environment and thus require preparation of a more detailed environmental impact statement. If the action will not significantly affect the environment, then a Finding of No Significant Impact is prepared.

Environmental Impact Statement (EIS) — A document required of Federal agencies by the National Environmental Policy Act for major proposals or legislation significantly affecting the environment. A tool for decision making, it describes the positive and negative effects of the undertaking and alternative actions.

Environmental Justice — The fair treatment of people of all races, cultures, incomes, and educational levels with respect to the development, implementation, and enforcement of environmental laws, regulations, and policies. Fair treatment implies that no population of people should be forced to shoulder a disproportionate share of the negative environmental impacts of pollution or environmental hazards due to a lack of political or economic influence.

Environmental Survey — A documented, multi disciplined assessment (with sampling and analysis) of a facility to determine environmental conditions and to identify environmental problems requiring corrective action.

Epidemiology — The science concerned with the study of events that determine and influence the frequency and distribution of disease, injury, and other health-related events and their causes in a defined human population.

Equivalent Sound (Pressure) Level — The equivalent steady sound level that, if continuous during a specified time period, would contain the same total energy as the actual time varying sound. For example, L_{eq} (1-h) and L_{eq} (24-h) are the 1-hour and 24-hour equivalent sound levels, respectively.

Existing Facilities — Facilities that are projected to exist as of the Record of Decision for this environmental impact statement, scheduled for July 2000.

Exposure Limit — The level of exposure to a hazardous chemical (set by law or a standard) at which or below which adverse human health effects are not expected to occur:

(1) Reference dose is the chronic exposure dose (milligrams or kilograms per day) for a given hazardous chemical at which or below which adverse human noncancer health effects are not expected to occur.

(2) Reference concentration is the chronic exposure concentration (milligrams per cubic meter) for a given hazardous chemical at which or below which adverse human noncancer health effects are not expected to occur.

External Accident — Accidents initiated by manmade energy sources not associated with operation of a given facility. Examples include airplane crashes, induced fires, and transportation accidents adjacent to a facility.

Fault — A fracture or a zone of fractures within a rock formation along which vertical, horizontal, or transverse slippage has occurred. A normal fault occurs when the hanging wall has been depressed in relation to the footwall. A reverse fault occurs when the hanging wall has been raised in relation to the footwall.

Finding of No Significant Impact — A document by a Federal agency briefly presenting the reasons why an action, not otherwise excluded, would not have a significant effect on the human environment and would not require an environmental impact statement under the National Environmental Policy Act.

Fissile Materials — Although sometimes used as a synonym for fissionable material, this term has acquired a more restricted meaning, namely, any material fissionable by thermal (slow) neutrons. The three primary fissile materials are uranium-233, uranium-235, and plutonium-239.

Fission (Fissioning) — The splitting 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 — Elements formed by the fission of heavy elements (primary fission products); also, the elements formed by the decay of the primary fission products, many of which are radioactive.

Fissionable Material — Material that could undergo fission by the absorption of fast neutrons.

Fissium — An alloy of molybdenum, ruthenium, rhodium, palladium, zirconium, and niobium that is designed to simulate fission products.

Floodplain — The lowlands adjoining inland and coastal waters and relatively flat areas.

Formation — In geology, the primary unit of formal stratigraphic mapping or description. Most formations possess certain distinctive features.

Fossil — Impression of trace of an animal or plant of past geological ages that has been preserved in the earth's crust.

Fuel Assembly — A cluster of fuel elements (or rods).

Fuel Element — Nuclear reactor component that includes the fissile material (fuel pin) sealed in cladding.

Fuel Pin — The uranium metal or alloy that undergoes fission in a nuclear reactor.

Fugitive Emissions — Emissions to the atmosphere from pumps, valves, flanges, seals, and other process points not vented through a stack. Also includes emissions from area sources such as ponds, lagoons, landfills, piles of stored material, and exposed soil.

g — A designator for ground motion acceleration, the rate of displacement of the ground due to the passage of elastic waves arising from earthquakes, explosions, seismic shots, machinery, wind, traffic, and other causes. The unit of acceleration is equal to about 9.8 meters (32.2 feet) per second^2.

Gamma-emitter — A radioactive substance that decays by releasing gamma radiation.

Gamma Rays — High-energy, short-wavelength, electromagnetic radiation accompanying fission and either emitted from the nucleus of an atom or emitted by some radionuclide or fission product. Gamma rays are very penetrating and can be stopped only by dense materials (such as lead) or a thick layer of shielding materials.

Gaussian Plume — The distribution of material (a plume) in the atmosphere resulting from the release of pollutants from a stack or other source. The distribution of concentrations about the centerline of the plume, which is assumed to decrease as a function of its distance from the source and centerline (Gaussian distribution), depends on the mean wind speed and atmospheric stability.

Genetic Effects — The outcome resulting from exposure to mutagenic chemicals or radiation which results in genetic changes in germ line or somatic cells. There are two kinds of genetic effects:

(1) Effects on genetic material in reproductive cells that cause trait modifications that can be passed from parents to offspring.

(2) Effects on genetic material in nonreproductive cells that result in tissue or organ modifications (e.g., liver tumors) that do not pass from parents to offspring.

Geologic Repository — A system that is intended to be used for, or may be used for, the disposal of radioactive waste or spent nuclear fuel in excavated geologic media. A geologic repository includes (1) the geologic repository operations area, and (2) the portion of the geologic setting that provides isolation. A near-surface disposal area is not a geologic repository.

Geology — The science that deals with the Earth: the materials, processes, environments, and history of the planet, including the rocks and their formation and structure.

Groundwater — The supply of water found beneath the Earth's surface, usually in aquifers, which may supply wells and springs.

Habitat — The environment occupied by individuals of a particular species, population, or community.

Half-Life — The time in which half the atoms of a radioactive isotope decay to another nuclear form. Half-lives vary from millionths of a second to billions of years.

Hazardous Chemical — Under 29 CFR 1910, Subpart Z, "hazardous chemicals" are defined as "any chemical which is a physical hazard or a health hazard." Physical hazards include combustible liquids, compressed gases, explosives, flammables, organic peroxides, oxidizers, pyrophorics, and reactives. A health hazard is any chemical for which there is good evidence that acute or chronic health effects occur in exposed employees. Hazardous chemicals include carcinogens, toxic or highly toxic agents, reproductive toxins, irritants, corrosives, sensitizers, hepatotoxins, nephrotoxins, agents that act on the hematopoietic system, and agents that damage the lungs, skin, eyes, or mucous membranes.

Hazardous Material — A material, including a hazardous substance, as defined by 49 CFR 171.8, which poses a risk to health, safety, and property when transported or handled.

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/Toxic Air Pollutants — Air pollutants known or suspected to cause serious health problems such as cancer, poisoning, or sickness, and may have immunological, neurological, reproductive, developmental, or respiratory effects.

Hazardous Waste — Any solid waste (can also be semisolid or liquid, or contain gaseous material) having the characteristics of ignitability, corrosivity, toxicity, or reactivity, defined by the Resource Conservation and Recovery Act and identified or listed in 40 CFR 261 or by the Toxic Substances Control Act.

Heavy Metals — Metallic or semimetallic elements of high molecular weight, such as mercury, chromium, cadmium, lead, and arsenic, that are toxic to plants and animals at known concentrations.

High Efficiency Particulate Air Filter — A filter used to remove very small particulates from dry gaseous effluent streams.

High-Level Radioactive Waste — The (1) highly radioactive waste material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid waste derived from such liquid waste that contains fission products in sufficient concentrations; and (2) other highly radioactive material that is determined, consistent with existing law, to require permanent isolation.

Historic Resources — Archaeological sites, architectural structures, and objects produced after the advent of written history dating to the time of the first Euro-American contact in an area.

Hot Cell/Hot Cell Facility — A heavily shielded enclosure for handling and processing (by remote means or automatically), or storing highly radioactive materials.

Impingement — The process by which aquatic organisms too large to pass through the screens of a water intake structure become caught on the screens and are unable to escape.

Inert cell — An enclosure where operations that require very low oxygen levels are performed.

Ingot — A mass of metal cast in a standard shape for convenient storage or shipment.

Involved Worker — Workers that would be involved in a proposed action as opposed to workers that would be on the site of a proposed action but not involved in the action.

Ionizing Radiation — Alpha particles, beta particles, gamma rays, neutrons, high-speed electrons, high-speed protons, and other particles or electromagnetic radiation that can displace electrons from atoms or molecules, thereby producing ions.

Isotope — An atom of a chemical element with a specific atomic number and atomic mass. Isotopes of the same element have the same number of protons, but different numbers of neutrons and different atomic masses.

Joule — A metric unit of energy, work, or heat, equivalent to 1 watt-second, 0.737 foot-pound, or 0.239 calories.

Karst Terrain — A type of land surface that is found in regions underlain by soluble rocks, such as limestone and dolomite, which is peculiar to dependent upon underground solution of the bedrock and the diversion of the surface waters to underground waters (that is, stream that disappear underground). Karst terrain is characterized by sinkholes, underground streams, and caves.

Landscape Character — The arrangement of a particular landscape as formed by the variety and intensity of the landscape features (land, water, vegetation, and structures) and the four basic elements (form, line, color, and texture). These factors give an area a distinctive quality that distinguishes it from its immediate surroundings.

Latent Cancer Fatalities — Fatalities associated with acute or chronic environmental exposure to chemicals or radiation that occur from delayed effects years after exposure.

Liquid Metal Cooled Breeder Reactor — A reactor that creates more fissionable material than it consumes and uses liquid metal as a coolant. Liquid sodium is a common metal used to cool this type of reactor.

Long-Lived Isotopes — Radionuclides with half-lives greater than about 30 years.

Long-term Storage — The storage of hazardous waste (a) onsite (a generator site) for a period of 90-days or greater, other than in a satellite accumulation area, or (b) offsite in a properly managed treatment, storage, or disposal facility for any period of time.

Low-Level Radioactive Waste — Waste that contains radioactivity, but is not classified as high-level radioactive waste, transuranic waste, spent nuclear fuel, or by-product material as defined by Section 11e (2) of the Atomic Energy Act of 1954, as amended.

Management — As used in this EIS, the stabilization and interim storage of sodium-bonded spent nuclear fuel pending final disposition.

Maximum Contaminant Level — The maximum permissible level of a contaminant in water delivered to any user of a public drinking water system. Maximum contaminant levels are enforceable standards under the Safe Drinking Water Act.

Maximally Exposed Individual (MEI) — A hypothetical individual defined to allow dose or dosage comparison with numerical criteria for the public. This individual is located at the point on the U.S. Department of Energy site boundary nearest to the facility in question. A hypothetical person who could potentially receive the maximum dose of radiation or hazardous chemicals.

Megajoule — A unit of heat, work, or energy equal to 1 million joules. See "Joule."

Meteorology — The science dealing with the atmosphere and its phenomena, especially as relating to weather.

Metric Tons of Heavy Metal — Quantities of unirradiated and spent nuclear fuel 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.

Migration — The natural movement of a material through the air, soil, or groundwater; also, seasonal movement of animals from one area to another.

Millirem — One thousandth of a rem.

Mixed Waste — Waste that contains both a hazardous waste subject to Resource Conservation and Recovery Act, and source, special nuclear or by-product material subject to the Atomic Energy Act of 1954 (42 U.S.C. 2011 *et seq.*).

Mollusks — Unsegmented, invertebrate animals including gastropods, pelecypods, and cephalopods.

National Ambient Air Quality Standards (NAAQS) — Uniform, national air quality standards established by the Environmental Protection Agency under the authority of the Clean Air Act that restrict ambient levels of criteria pollutants to protect public health (primary standards) or public welfare (secondary standards), including plant and animal life, visibility, and materials. Standards have been set for ozone, carbon monoxide, particulates, sulfur dioxide, nitrogen, nitrogen dioxide, and lead.

National Emission Standards for Hazardous Air Pollutants — A set of national emission standards for listed hazardous pollutants emitted from specific classes or categories of new and existing sources.

National Pollutant Discharge Elimination System (NPDES) — Federal permitting system required for water pollution effluent under the Clean Water Act, as amended.

National Register of Historic Places — A list maintained by the Secretary of the Interior of districts, sites, buildings, structures, and objects of prehistoric or historic local, state, or national significance under Section 2(b) of the Historic Sites Act of 1935 (16 U.S.C. 462) and Section 101(a) (1) (A) of the National Historic Preservation Act of 1966, as amended.

Neutron — An uncharged elementary particle with a mass slightly greater than that of the proton, found in the nucleus of every atom heavier than hydrogen-1. A free neutron is unstable and decays with a half-life of about 13 minutes into an electron and a proton; used in the fission process.

Neutron Flux — The product of neutron number density and velocity (energy), giving an apparent number of neutrons flowing through a unit area per unit time.

Neutron Poison — A material (e.g., a boron or a burnable absorber rod) inserted into a nuclear reactor or spent fuel pool to absorb neutrons and end criticality. Any material with a strong affinity for absorbing neutrons without generating new neutrons that can be used to control the nuclear chain reaction.

Nitrogen Oxides — Refers to the oxides of nitrogen, primarily nitrogen oxide (NO) and nitrogen dioxide (NO_2). These are produced in the combustion of fossil fuel and can constitute an air pollution problem. Nitrogen dioxide emissions contribute to acid deposition and formation of atmospheric ozone.

Noble metals — A group of metals such as zirconium, niobium, and gold that are highly resistant to oxidation and corrosion.

Noise — Any sound that is undesirable because it interferes with speech and hearing, or is intense enough to damage hearing, or is otherwise annoying (unwanted sound).

Nonattainment Area — An air quality control region (or portion thereof) in which the Environmental Protection Agency has determined that ambient air concentrations exceed national ambient air quality standards for one or more criteria pollutants.

Normal Conditions — All activities associated with a facility mission, whether operation, maintenance, storage, and so forth, which are carried out within a defined envelope. This envelope can be design process conditions, performance in accordance with procedures, and so forth.

Notice of Intent — Announces the scoping process. The Notice of Intent is usually published in the Federal Register and a local newspaper. The scoping process includes holding at least one public meeting and requesting written comments on what issues and environmental concerns an environmental impact statement should address.

Nuclear Power Plant — A facility that converts nuclear energy into electrical power.

Nuclear Radiation — Particles (alpha, beta, neutrons) or photons (gamma) emitted from the nucleus of unstable radioactive atoms as a result of radioactive decay.

Nuclear Reaction — A reaction in which an atomic nucleus is transformed into another isotope of that respective nuclide, or into another element altogether; it is always accompanied by the liberation of either particles or energy.

Nuclear Reactor — A device that sustains a controlled nuclear fission chain reaction that releases energy in the form of heat.

Nuclear Regulatory Commission (NRC) — The Federal agency that regulates the civilian nuclear power industry in the United States.

Nuclide — A species of atom characterized by the constitution of its nucleus and, hence, by the number of protons, the number of neutrons, and the energy content.

Obsidian — Volcanic glass, which may be banded and is usually black in color, although examples in red, green, and brown are also known.

Occupational Safety and Health Administration (OSHA) — A Federal agency that oversees and regulates workplace health and safety; created by the Occupational Safety and Health Act of 1970.

Off-gas — A volatile and semivolatile gaseous product that is released during a process.

Off Site — As used in the environmental impact statement, the term denotes a location, facility, or activity occurring outside of the boundary of the facility of interest.

Ozone — The triatomic form of oxygen; in the stratosphere, ozone protects the Earth from the sun's ultraviolet rays, but in lower levels of the atmosphere, ozone is considered an air pollutant.

Packaging — With regard to hazardous or radionuclide materials, the assembly of components necessary to ensure compliance with Federal regulations for transportation. It may consist of one or more receptacles, absorbent materials, spacing structures, thermal insulation, radiation shielding, and devices for cooling or absorbing mechanical shocks. The vehicle tie-down system and auxiliary equipment may be designated as part of the packaging.

Particulate Matter — Air pollutants including dust, dirt, soot, smoke, or liquid droplets emitted into the air. "Total suspended particulate" was first used as the indicator for particulate concentrations. Current standards use the indicators "PM_{10}" and "$PM_{2.5}$," which include only those particles with an aerodynamic diameter smaller than or equal to 10 micrometers and 2.5 micrometers, respectively. The smaller particles are more responsible for adverse health effects because they reach further into the respiratory tract.

Permutation — Changing the order of elements arranged in a particular order.

Person-Rem — The unit of collective radiation dose to a given population; the sum of the individual doses received by a population segment.

Playa — A dry lake bed in a desert basin or a closed depression that contains water on a seasonal basis.

Plume — A flowing, often somewhat conical, trail of emissions from a continuous point source.

Plume Immersion — With regard to radiation, the situation in which an individual is enveloped by a cloud of radiation gaseous effluent and receives an external radiation dose.

Plutonium — A heavy, radioactive, metallic element with the atomic number 94. It is produced artificially in a reactor by bombardment of uranium with neutrons and is used in the production of nuclear weapons.

Poison — See "neutron poison."

Pounds per Square Inch — A measure of pressure; atmospheric pressure is about 14.7 pounds per square inch.

Prevention of Significant Deterioration — An Environmental Protection Agency program, mandated by the Clean Air Act, in which state or Federal permits are required that are intended to limit increases in air pollutant concentrations by restricting emissions for new or modified sources in places where air quality is already better than required to meet primary and secondary ambient air quality standards.

Prime Farmland — Land that has the best combination of physical and chemical characteristics for producing food, feed, fiber, forage, oil-seed, and other agricultural crops with minimum inputs of fuel, fertilizer, pesticides, and labor without intolerable soil erosion, as determined by the Secretary of Agriculture (Farmland Protection Act of 1981, 7 CFR 7, paragraph 658).

Probabilistic Risk Assessment — A comprehensive, logical, and structured methodology to identify and quantitatively evaluate significant accident sequences and their consequences.

Probable Maximum Flood — Flood levels predicted for a scenario having hydrological conditions that maximize the flow of surface waters.

Programmatic Environmental Impact Statement — A legal document prepared in accordance with the requirements of Section 102(2)(C) of the National Environmental Policy Act which evaluates the environmental impacts of proposed Federal actions that involve multiple decisions potentially affecting the environment at one or more sites.

Proliferation (Nuclear) — The spread of nuclear weapons and the materials and technologies used to produce them.

PUREX (Plutonium Uranium Extraction) — A chemical separation process that has been used for recovering uranium and plutonium from irradiated fuel in a form usable as reactor fuel or for weapons. The process uses aqueous solvent extraction to perform the separation. This technology can also be used to treat spent nuclear fuel for disposal.

Pyrophoric — Being highly susceptible to spontaneous ignition and continuous combustion.

Qualitative Environmental Impacts — 10 CFR 51, Appendix B defines the qualitative terms "small," "moderate," and "large" as follows:

Small	Environmental effects are not detectable or are so minor that they would neither destabilize nor noticeably alter any important attribute of the resource. For the purposes of assessing radiological impacts, the U.S. Nuclear Regulatory Commission (NRC) has concluded that those impacts that do not exceed permissible levels in the NRC's regulations are considered small.
Moderate	Environmental effects are sufficient to alter noticeably, but not to destabilize, important attributes of the resource.
Large	Environmental effects are clearly noticeable and are sufficient to destabilize important attributes of the resource.

Quality Factor — The principal modifying factor that is employed to derive dose equivalent from absorbed dose.

Rad — See "radiation absorbed dose."

Radiation — The emitted particles or photons from the nuclei of radioactive atoms. Some elements are naturally radioactive; others are induced to become radioactive by bombardment in a reactor.

Radiation Absorbed Dose (rad) — The basic unit of absorbed dose equal to the absorption of 0.01 Joule per kilogram of absorbing material.

Radioactive Mixed Waste — Waste containing both radioactive and hazardous components regulated by the Atomic Energy Act and the Resource Conservation and Recovery Act, respectively. The term "radioactive component: refers only to the actual radionuclides dispersed or suspended in the waste substance.

Radioactive Waste — Materials from nuclear operations that are radioactive or are contaminated with radioactive materials, and for which use, reuse, or recovery are impractical.

Radioactivity — The spontaneous decay or disintegration of unstable atomic nuclei, accompanied by the emission of radiation.

Radioisotopes — Radioactive nuclides of the same element (same number of protons in their nuclei) that differ in the number of neutrons.

Radionuclide — A radioactive element characterized according to its atomic mass and atomic number which can be man-made or naturally occurring.

Radon — Gaseous, radioactive element with the atomic number 86 resulting from the radioactive decay or radium. Radon occurs naturally in the environment, and can collect in unventilated enclosed areas, such as basements. Large concentrations of radon can cause lung cancer in humans.

RADTRAN — A computer code that combines user-determined meteorological, demographic, transportation, packaging, and material factors with health physics data to calculate the expected radiological consequences and accident risk of transporting radioactive material.

Reactive — Having low chemical stability and subject to high chemical reaction rates.

Record of Decision — A document prepared in accordance with the requirements of the Council on Environmental Quality and National Environmental Policy Act regulations 40 CFR 1505.2, that provides a concise public record of the decision on a proposed Federal action for which an environmental impact statement was prepared. A Record of Decision identifies the alternatives considered in reaching the decision, the environmentally preferable alternative(s), factors balanced in making the decision, whether all practicable means to avoid or minimize environmental harm have been adopted, and if not, why they were not.

Regional Economic Area — A geographic area consisting of an economic node and the surrounding counties that are economically related and include the places of work and residences of the labor force. Each regional economic area is defined by the U.S. Bureau of Economic Analysis.

Region of Influence — A site-specific geographic area that includes the counties where approximately 90 percent of the current U.S. Department of Energy and/or contractor employees reside.

Rem — See "roentgen equivalent man."

Remediation — The process, or a phase in the process, of rendering radioactive, hazardous, or mixed waste environmentally safe, whether through processing, entombment, or other methods.

Reprocessing (of spent nuclear fuel) — Processing of reactor-irradiated nuclear material (primarily spent nuclear fuel) to recover fissile and fertile material, to recycle such materials primarily for defense programs. Historically, reprocessing has involved aqueous chemical separation of elements (typically uranium or plutonium) from undesired elements in the fuel.

Riparian — Of, on, or relating to the banks of a natural course of water.

Risk — A quantitative or qualitative expression of possible loss that considers both the probability that a hazard will cause harm and the consequences of that event.

Risk Assessment (chemical or radiological) — The qualitative and quantitative evaluation performed in an effort to define the risk posed to human health and/or the environment by the presence or potential presence and/or use of specific chemical or radiological materials.

Roentgen — A unit of exposure to ionizing X or gamma radiation equal to or producing 1 electrostatic unit of charge per cubic centimeter of air. It is approximately equal to 1 rad.

Roentgen Equivalent Man (rem) — A measure of radiation dose (i.e., the average background radiation dose is 0.3 rem per year). The unit of biological dose equal to the product of the absorbed dose in rads; a quality factor, which accounts for the variation in biological effectiveness of different types of radiation; and other modifying factors.

Runoff — The portion of rainfall, melted snow, or irrigation water that flows across the ground surface and eventually enters streams.

Safety Analysis Report — A document that provides a complete description and safety analysis of a facility design, normal and emergency operations, hypothetical accidents and their predicted consequences, and the means proposed to prevent such accidents or mitigate their consequences.

Safety Evaluation Report — A document prepared by the U.S. Nuclear Regulatory Commission that evaluates documentation (i.e., technical specifications, safety analysis reports, and special safety reviews and studies) submitted by a licensee for its approval. This ensures that all of the safety aspects of part or all of the activities conducted at the facility are formally and thoroughly analyzed, evaluated, and recorded.

Sanitary waste — Waste generated by normal housekeeping activities, liquid or solid (including sludge), which are not hazardous or radioactive.

Scope — In a document prepared pursuant to the National Environmental Policy Act of 1969, the range of actions, alternatives, and impacts to be considered.

Scoping — The solicitation of comments from interested persons, groups, and agencies at public meetings, public workshops, in writing, electronically, or via fax to assist in defining the proposed action, identifying alternatives, and developing preliminary issues to be addressed in an environmental impact statement.

Seismic — Pertaining to any Earth vibration, especially an earthquake.

Seismic Zone — An area defined by the Uniform Building Code (1991), designating the amount of damage to be expected as the result of earthquakes. The United States is divided into six zones: (1) Zone 0: no damage; (2) Zone 1: minor damage, corresponds to intensities V and VI of the modified Mercalli intensity scale; (3) Zone 2A: moderate damage, corresponds to intensity VII of the modified Mercalli intensity scale (eastern U.S.); (4) Zone 2B: slightly more damage than 2A (western U.S.); (5) Zone 3: major damage, corresponds to intensity VII and higher of the modified Mercalli intensity scale; (6) Zone 4: areas within Zone 3 determined by proximity to certain major fault systems.

Severe Accident — An accident with a frequency rate of less than 10^{-6} per year that would have more severe consequences than a design-basis accident, in terms of damage to the facility, offsite consequences, or both.

Sewage — The total of organic waste and wastewater generated by an industrial establishment or a community.

Shielding — With regard to radiation, any material of obstruction (bulkheads, walls, or other construction) that absorbs radiation in order to protect personnel or equipment.

Short-Lived Nuclides — Radioactive isotopes with half-lives no greater than about 30 years (e.g., cesium-137 and strontium-90).

Shutdown — For a U.S. Department of Energy (DOE) reactor, that condition in which the reactor has ceased operation and DOE has declared officially that it does not intend to operate it further (see DOE Order 5480.6, *Safety of Department of Energy-Owned Nuclear Reactors*).

Silt — A sedimentary material consisting of fine mineral particles intermediate in size between sand and clay.

Sinkhole — A depression in the earth's surface formed by the collapse of a cavern roof. Typically associated with Karst terrain.

Sodium-bonded — Physically in contact with and attached to the element sodium.

Source Term — The estimated quantities of radionuclides or chemical pollutants available for release to the environment.

Species of Special Concern — Native species that are either low in numbers, limited in distribution, or have suffered significant habitat losses.

Spent Nuclear Fuel — Fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated for reprocessing.

Standardized Canister —As used in this EIS, this refers to a standard-sized U.S. Department of Energy canister which is stainless steel, right circular cylinder with a nominal outside diameter of 45.7 centimeters (18 inches), or 61 centimeters (24 inches), a nominal thickness of 0.95 centimeters (0.375 inches), or 1.27 centimeters (0.50 inches), respectively. Each canister could have a maximum overall length of either 3 meters (118.11 inches) with a useable length of 2.25 meters (88.58 inches), or 4.57 meters (179.92 inches) with useable length of 4.11 meters (161.81 inches). The standardized canister is used for storing spent nuclear fuel assemblies, high-integrity cans, and any other waste packages.

Surface Water — Water on the Earth's surface, as distinguished from water in the ground (groundwater).

Threatened Species — Any species designated under the Endangered Species Act as likely to become an endangered species within the foreseeable future throughout all or a significant portion of its range.

Threshold Limit Values — The recommended highest concentrations of contaminants to which workers may be exposed according to the American Conference of Governmental Industrial Hygienists.

Transuranic Waste — Waste contaminated with alpha-emitting radionuclides with half-lives greater than 20 years and concentrations greater than 100 nanocuries/gram at time of assay. A nanocurie is 10^{-9} curies.

Treatment — In this EIS, a process to remove and/or stabilize metallic sodium.

Unusual Occurrence — Any unusual or unplanned event that adversely affects or potentially affects the performance, reliability, or safety of a facility.

Uranium — A heavy, silvery-white metallic element (atomic number 92) with several radioactive isotopes that is used as fuel in nuclear reactors or as radiation shielding.

Viewshed — The extent of an area that may be viewed from a particular location. Viewsheds are generally bounded by topographic features such as hills or mountains.

Visual Resource Management Class — A class defines the different degrees of modification allowed to the basic elements of landscape. They are: Class I – Applied to wilderness areas, wild and scenic rivers, and other similar situations - the level of change to the characteristic landscape should be very low and must not attract attention; Class II – Management activities may be seen, but do not attract attention; Class III – Management activities may attract attention, but should not dominate the view; Class IV – Management activities may dominate the view.

Vitreous — Resembling or having the nature of glass.

Vitrification — The process of immobilizing waste material that results in glass-like solid.

Volatile Organic Compounds — A broad range of organic compounds that vaporize at ambient or relatively low temperatures, such as benzene, chloroform, and methyl alcohol. With regard to air pollution, any organic compound that participates in atmospheric photochemical reaction, except for those designated by the Environmental Protection Agency administrator as having negligible photochemical reactivity.

Waste Minimization and Pollution Prevention — An action that economically avoids or reduces the generation of waste and pollution by source reduction, reducing the toxicity of hazardous waste and pollution, improving energy use, or recycling. These actions will be consistent with the general goal of minimizing present and future threats to human health, safety, and the environment.

Weighting Factor — With regard to radiation, the fraction of the total health risk resulting from uniform whole-body irradiation that could be contributed to that particular tissue.

Wetlands — Those areas that are inundated or saturated by surface or ground water at a frequency and duration sufficient to support, and that under normal circumstances do support, a prevalence of vegetation typically adapted for life in saturated soil conditions. Wetlands generally include swamps, marshes, bogs, and similar areas.

Whole-Body Dose — With regard to radiation, the dose resulting from the uniform exposure of all organs and tissues in a human body. (Also see "effective dose equivalent.")

Wind Rose — A depiction of wind speed and direction frequency for a given period of time.

X/Q (Chi/Q) — The relative calculated air concentration due to a specific air release and atmospheric dispersion; units are (seconds per cubic meter). For example (curies per cubic meter)/(curies per second) = (seconds per cubic meter) or (grams per cubic meter)/(grams per second) = (seconds per cubic meter).

Zeolite — One of approximately 30 hydrous (water containing) aluminum silicate minerals or their corresponding synthetic compounds, used chiefly as molecular filters and ion-exchange agents such as is used in a water softener. It is used in electrometallurgical treatment to collect and contain fission products from process salt.

Chapter 7

List of Preparers

7. LIST OF PREPARERS

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B.A., English, DePauw University

Experience/ Technical Specialty: Twelve years. NEPA compliance, regulatory compliance and permitting, socioeconomics, environmental justice, comprehensive land use and development planning, and cultural resources.

ROBERT SCHLEGEL, SCIENCE APPLICATIONS INTERNATIONAL CORPORATION
EIS RESPONSIBILITIES: RADIOLOGICAL IMPACT ASSESSMENTS AND ACCIDENTS

Education: Degree of Nuclear Engineer, Columbia University
M.S., Nuclear Engineering, Columbia University
B.S., Chemical Engineering, Massachusetts Institute of Technology

Experience/ Technical Specialty: Thirty-four years. Radiological impact assessments, radiological dose/health effects calculations, and safety analysis.

EDWARD Y. SHUM, SCIENCE APPLICATIONS INTERNATIONAL CORPORATION
EIS RESPONSIBILITIES: TECHNICAL REVIEWER AND PREPARER OF RESPONSES TO PUBLIC COMMENTS

Education: Ph.D., Environmental and Nuclear Chemistry, Oregon State University
M.S., Nuclear Chemistry, Oregon State University
B.S., Chemistry, University of California, Berkeley

Experience/ Technical Specialty: Twenty-six years in licensing nuclear facilities spent nuclear fuel, senior safety and environmental project manager, health physics, dose assessment, and decommissioning and emergency planning.

CARL A. SNYDER, SCIENCE APPLICATIONS INTERNATIONAL CORPORATION
EIS RESPONSIBILITIES: RADIOLOGICAL ASSESSMENT ANALYSIS

Education: B.S., Nuclear Engineering, University of Maryland
B.S., Mathematics, University of Maryland

Experience/ Technical Specialty: Six years. Nuclear safety, nuclear criticality, thermal hydraulics, licensing, accident analysis, and dose consequences.

CATHY STEWART, SCIENCE APPLICATIONS INTERNATIONAL CORPORATION
EIS RESPONSIBILITIES: ADMINISTRATIVE RECORD

Education: Computer Information Systems course work, Strayer College

Experience/ Technical Specialty: Twenty years. Database and records management, and congressional affairs.

GILBERT H. WALDMAN, SCIENCE APPLICATIONS INTERNATIONAL CORPORATION
EIS RESPONSIBILITIES: RADIOLOGICAL ASSESSMENT ANALYSIS

Education: M.S., Technical Management, Johns Hopkins University
B.S., Nuclear Engineering, University of Florida

Experience/ Technical Specialty: Eight years. Radiological impacts analysis, radiological dose modeling, and radiological risk assessments.

ROBERT H. WERTH, SCIENCE APPLICATIONS INTERNATIONAL CORPORATION
EIS RESPONSIBILITIES: NONRADIOLOGICAL ASSESSMENT ANALYSIS

Education: B.A., Physics, Gordon College

Experience/ Technical Specialty: Twenty-four years. Acoustics and air quality analysis, regulatory reviews, and NEPA documentation.

JOHN W. WILLIAMS, SCIENCE APPLICATIONS INTERNATIONAL CORPORATION
EIS RESPONSIBILITIES: ENVIRONMENTAL JUSTICE

Education: Ph.D., Physics, New Mexico State University
M.S., Physics, New Mexico State University
B.S., Mathematics, North Texas State University

Experience/ Technical Specialty: Twenty-three years. NEPA compliance, electromagnetic models, air quality modeling, and ionizing radiation impacts and safety.

Chapter 8

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Approximately 300 copies of the Final EIS were sent to stakeholders
Approximately 1,500 copies of the Summary of the Final EIS were sent to stakeholders

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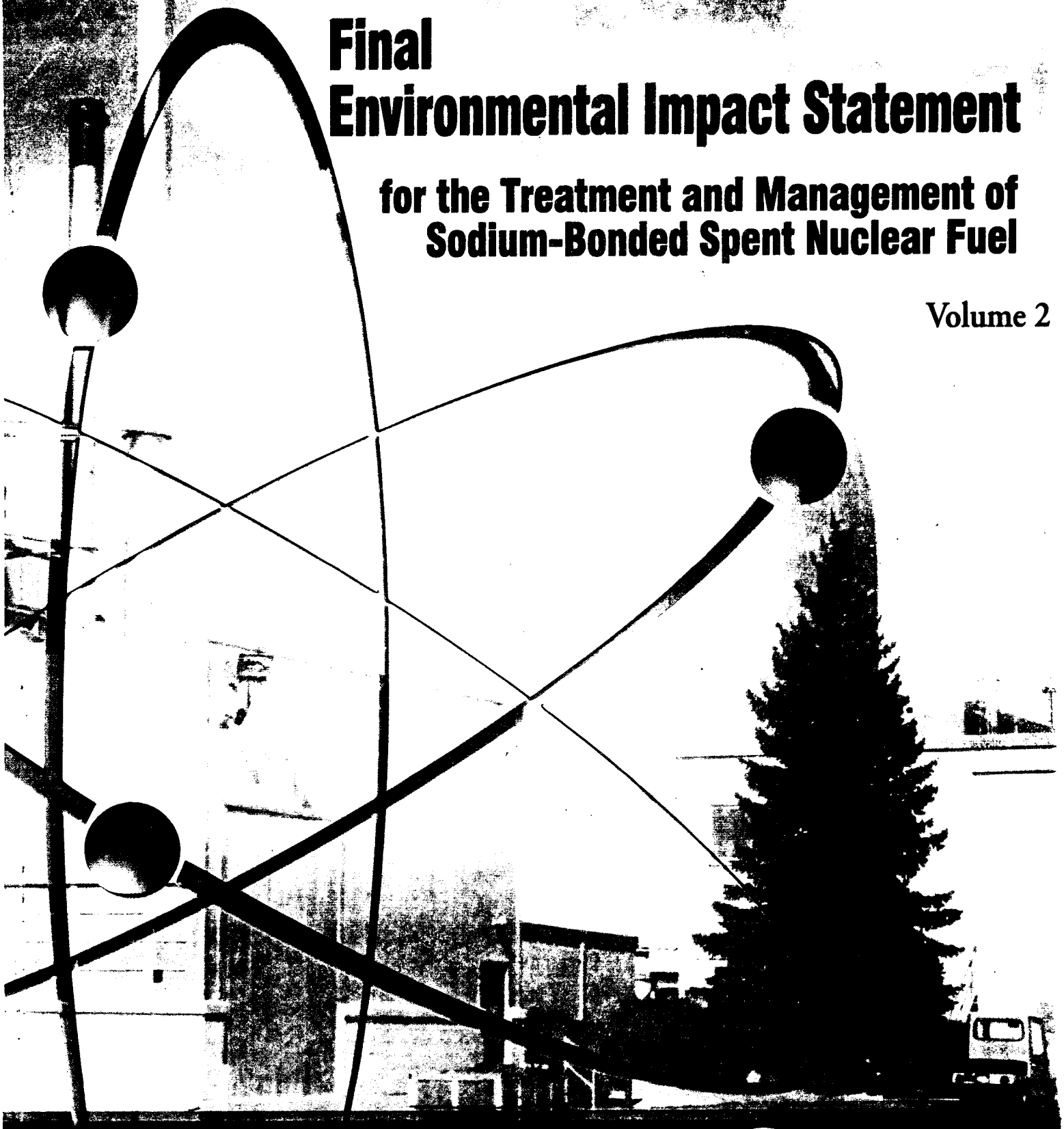
Final Environmental Impact Statement

for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel

Volume 2



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Abstract: DOE is responsible for the safe and efficient management of its sodium-bonded spent nuclear fuel. This fuel contains metallic sodium, a highly reactive material; metallic uranium, which is also reactive; and in some cases, highly enriched uranium. The presence of reactive materials could complicate the process of qualifying and licensing DOE's sodium-bonded spent nuclear fuel inventory for disposal in a geologic repository. Currently, more than 98 percent of this inventory is located at the Idaho National Engineering and Environmental Laboratory (INEEL), near Idaho Falls, Idaho. In addition, in a 1995 agreement with the State of Idaho, DOE committed to remove all spent nuclear fuel from Idaho by 2035. This EIS evaluates the potential environmental impacts associated with the treatment and management of sodium-bonded spent nuclear fuel in one or more facilities located at Argonne National Laboratory-West (ANL-W) at INEEL and either the F-Canyon or Building 105-L at the Savannah River Site (SRS) near Aiken, South Carolina. DOE has identified and assessed six proposed action alternatives in this EIS. These are: (1) electrometallurgical treatment of all fuel at ANL-W, (2) direct disposal of blanket fuel in high-integrity cans with the sodium removed at ANL-W, (3) plutonium-uranium extraction (PUREX) processing of blanket fuel at SRS, (4) melt and dilute processing of blanket fuel at ANL-W, (5) melt and dilute processing of blanket fuel at SRS, and (6) melt and dilute processing of all fuel at ANL-W. In addition, Alternatives 2 through 5 include the electrometallurgical treatment of driver fuel at ANL-W. Under the No Action Alternative, the EIS evaluates both the continued storage of sodium-bonded spent nuclear fuel until the development of a new treatment technology or direct disposal without treatment. Under all of the alternatives, the affected environment is primarily within 80 kilometers (50 miles) of spent nuclear fuel treatment facilities. Analyses indicate little difference in the environmental impacts among alternatives. DOE has identified electrometallurgical treatment as its Preferred Alternative for the treatment and management of all sodium-bonded spent nuclear fuel, except for the Fermi-1 blanket fuel. The No Action Alternative is preferred for the Fermi-1 blanket spent nuclear fuel.

Public Comments: The draft EIS was issued for public review and comment on July 31, 1999. The comment period ended on September 28, 1999, although late comments were accepted. Public hearings to solicit

comments on the draft EIS were held in North Augusta, South Carolina; Boise and Idaho Falls, Idaho; and Arlington, Virginia. All comments were considered during the preparation of the final EIS, which also incorporates additional and new information received since the issuance of the draft EIS. In response to comments on the SBSNF Draft EIS and as a result of information that was unavailable at the time of the issuance of the draft EIS, the final EIS contains revisions and new information. These revisions and new information are indicated by a double underline for minor word changes or by a sidebar in the margin for sentence or larger additions. Appendix A contains the comments received during the public review period of the SBSNF Draft EIS and DOE's responses to these comments. DOE will use the analyses presented in this final EIS as well as other information in preparing the Record of Decision for the treatment and management of its sodium-bonded spent nuclear fuel. DOE will issue this Record of Decision no sooner than 30 days after the U.S. Environmental Protection Agency publishes a notice of availability of this final EIS in the *Federal Register*.

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Appendix I

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Acronyms, Abbreviations, and Conversion Charts

ACRONYMS, ABBREVIATIONS, AND CONVERSION CHARTS

ANL-W	Argonne National Laboratory-West
BEIR	Biological Effects of Ionizing Radiation
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CFR	Code of Federal Regulations
CPP	Chemical Processing Plant
DOE	U.S. Department of Energy
EBR-II	Experimental Breeder Reactor-II
EIS	Environmental Impact Statement
EPA	U.S. Environmental Protection Agency
ERPG	Emergency Response Planning Guideline
FR	*Federal Register*
GMODS	Glass Material Oxidation and Dissolution System
HFEF	Hot Fuel Examination Facility
INEEL	Idaho National Engineering and Environmental Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
MEDEC	melt, drain, evaporate, and calcine (ANL-W process)
NAAQS	National Ambient Air Quality Standards
NEPA	National Environmental Policy Act
NPDES	National Pollutant Discharge Elimination System
NRC	U.S. Nuclear Regulatory Commission
OSHA	Occupational Safety and Health Administration
PM_n	Particulate matter less than or equal to *n* microns in diameter
PUREX	Plutonium-Uranium Extraction
RCRA	Resource Conservation and Recovery Act
SBSNF	Sodium-Bonded Spent Nuclear Fuel
SRS	Savannah River Site
U.S.C.	United States Code
YAG	yttrium-aluminum-garnet

Metric Conversion Chart

To Convert Into Metric			*To Convert From Metric*		
If You Know	**Multiply By**	**To Get**	**If You Know**	**Multiply By**	**To Get**
Length					
inches	2.54	centimeters	centimeters	0.3937	inches
feet	30.48	centimeters	centimeters	0.0328	feet
feet	0.3048	meters	meters	3.281	feet
yards	0.9144	meters	meters	1.0936	yards
miles	1.60934	kilometers	kilometers	0.6214	miles
Area					
square inches	6.4516	square centimeters	square centimeters	0.155	square inches
square feet	0.092903	square meters	square meters	10.7639	square feet
square yards	0.8361	square meters	square meters	1.196	square yards
acres	0.40469	hectares	hectares	2.471	acres
square miles	2.58999	square kilometers	square kilometers	0.3861	square miles
Volume					
fluid ounces	29.574	milliliters	milliliters	0.0338	fluid ounces
gallons	3.7854	liters	liters	0.26417	gallons
cubic feet	0.028317	cubic meters	cubic meters	35.315	cubic feet
cubic yards	0.76455	cubic meters	cubic meters	1.308	cubic yards
Weight					
ounces	28.3495	grams	grams	0.03527	ounces
pounds	0.4536	kilograms	kilograms	2.2046	pounds
short tons	0.90718	metric tons	metric tons	1.1023	short tons
Temperature					
Fahrenheit	Subtract 32, then multiply by 0.55556	Celsius	Celsius	Multiply by 1.8, then add 32	Fahrenheit

Metric Prefixes

Prefix	*Symbol*	*Multiplication Factor*
exa-	E	1 000 000 000 000 000 000 = 10^{18}
peta-	P	1 000 000 000 000 000 = 10^{15}
tera-	T	1 000 000 000 000 = 10^{12}
giga-	G	1 000 000 000 = 10^{9}
mega-	M	1 000 000 = 10^{6}
kilo-	k	1 000 = 10^{3}
hecto-	h	100 = 10^{2}
deka-	da	10 = 10^{1}
deci-	d	0.1 = 10^{-1}
centi-	c	0.01 = 10^{-2}
milli-	m	0.001 = 10^{-3}
micro-	μ	0.000 001 = 10^{-6}
nano-	n	0.000 000 001 = 10^{-9}
pico-	p	0.000 000 000 001 = 10^{-12}
femto-	f	0.000 000 000 000 001 = 10^{-15}
atto-	a	0.000 000 000 000 000 001 = 10^{-18}

Appendix A

Overview of the Public Participation Process

APPENDIX A
OVERVIEW OF THE PUBLIC PARTICIPATION PROCESS

This appendix describes the public comment process for the U.S. Department of Energy's *Draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel* and the procedures used to respond to those comments. Section A.1 provides an overview of the public scoping process for the draft environmental impact statement. Section A.2 discusses the process for obtaining public comments on the draft environmental impact statement, including the public hearing format and the major issues raised by the comments received. Section A.2.5 presents oral comments made by attendees at the four public hearings and the U.S. Department of Energy's responses. Section A.2.6 contains scanned copies of comment documents received during the public comment period and the Department's responses to each comment.

A.1 THE PUBLIC SCOPING PROCESS

A.1.1 Scoping Process Description

As a preliminary step in the development of an environmental impact statement (EIS), regulations established by the Council on Environmental Quality (40 CFR 1501.7) and the U.S. Department of Energy (DOE) require "an early and open process for determining the scope of issues to be addressed and for identifying the significant issues related to a proposed action." The purpose of this scoping process is: (1) to inform the public about a proposed action and the alternatives being considered and (2) to identify and/or clarify those issues considered most relevant by the public.

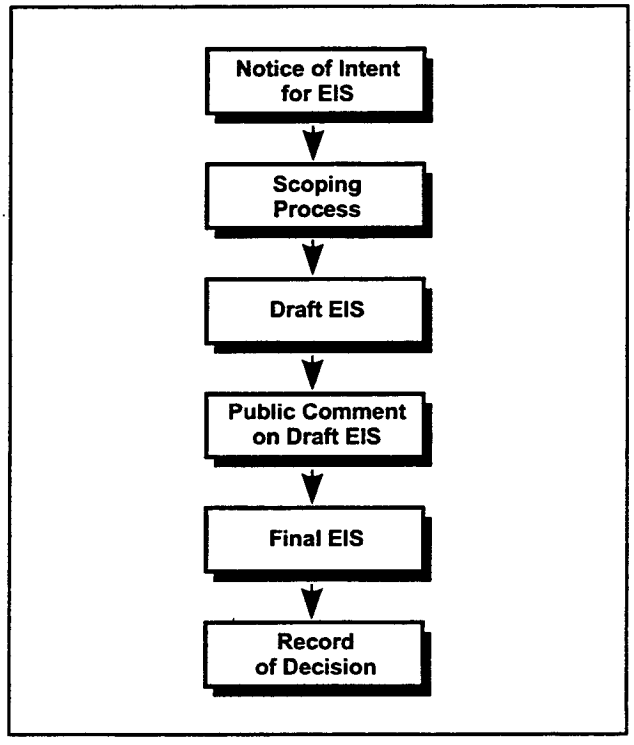

Figure A–1 NEPA Process

On February 22, 1999, DOE published in the *Federal Register* a Notice of Intent to prepare an EIS for the treatment of sodium-bonded spent nuclear fuel. As shown in **Figure A–1**, the scoping process is one of the opportunities for public involvement required as part of the National Environmental Policy Act (NEPA) process. The Notice of Intent listed the alternatives and issues initially identified by DOE for evaluation in the EIS. Members of the public, civic leaders, and other interested parties were invited to comment on these issues and to suggest additional issues that should be considered in the EIS. The Notice of Intent also informed the public that comments on the proposed action could be communicated via U.S. mail, a special DOE web site on the Internet, a toll-free phone line, a toll-free fax line, or in person at one of four public meetings.

Four public scoping meetings were held at locations in Idaho, South Carolina, and Virginia, near the Washington, DC, metropolitan area. The first public meeting was attended by about 60 members of the public and was held in Idaho Falls, Idaho, on March 9, 1999. The second meeting was held in Boise, Idaho, on

March 11, 1999, and was attended by about 7 members of the public. Approximately 10 members of the public attended the third meeting, which was held in North Augusta, South Carolina, on March 15, 1999. The fourth meeting was held in Arlington, Virginia, on March 18, 1999, and was attended by about 8 members of the public.

As a result of previous experience and positive responses from attendees of other DOE/NEPA public meetings and hearings, DOE chose an interactive format for the scoping meetings. Each meeting began with a presentation by a DOE representative who explained the proposed action. Afterwards, an impartial facilitator opened the floor to questions, comments, and concerns from the audience. DOE and national laboratory personnel were available to respond to the questions and comments as needed. A court reporter was provided at each of the meetings to record the oral comments, and personnel were available to receive any written statements or comments that were submitted at the meetings. In addition, the public was encouraged to submit written or verbal comments via letters, the DOE Internet web site, the toll-free phone line, or the toll-free fax line until the end of the scoping period on April 8, 1999 (45 days after publication of the Notice of Intent).

It should be noted that, for EIS public scoping purposes, a comment is defined as a single statement or opinion concerning a specific issue. Any statement may contain many separate comments. Most of the verbal and written public statements submitted during the EIS scoping period contained multiple comments on various individual issues.

A.1.2 Scoping Process Results

Two hundred twenty eight comments were received from citizens, interested groups, and other stakeholders during the public scoping comment period. Of these, 109 were verbal comments made during the public meetings. The remaining comments (119) either were submitted at the public meetings in written form or were received via mail, Internet, fax, or phone during the scoping comment period. In cases where a single commentor provided similar or identical comments both orally at the public meetings and in writing, each individual comment was counted once (i.e., repetitions were not counted).

Many members of the public who spoke at the public meetings asked specific, technical questions about the proposed action that were answered by the DOE and national laboratory representatives at each meeting. Primary areas of interest included:

- Waste volume reduction
- Nature of the spent nuclear fuel waste at Argonne National Laboratory-West (ANL-W)
- Waste forms characterization
- Waste disposition and qualification (repository acceptance criteria)
- Plutonium-uranium extraction (PUREX)
- Use of facilities
- Nonproliferation impacts
- Transportation
- Demonstration project

The comments obtained through the overall public scoping process addressed several key issues. A number of persons commented on the schedule for the EIS. Many said the draft EIS should not be issued for public comment before publication of other reports, such as the Waste Qualification Assessment from the National Research Council; the National Academy of Sciences' Independent Assessment Final Report on the demonstration project; a Nonproliferation Impacts Assessment by the DOE Office of Nonproliferation and National Security; and an independent study of the costs of the proposed action. Several commentors also said this EIS is premature because the demonstration project will not be completed until after the draft EIS is published.

Several commentors asked that the EIS include information about the costs of the proposed action and all of the technology alternatives under consideration. Other commentors stated the public should have an opportunity to comment on DOE's ongoing independent Nonproliferation Impacts Assessment within the same time frame as the draft EIS, or that this EIS should be delayed until the assessment is publicly available. Some suggested the assessment be included in the EIS. A few commentors expressed the opinion that electrometallurgical treatment of spent nuclear fuel is a proliferation-prone technology.

Waste was another issue that was frequently cited. Many waste-related comments included opinions about whether low-enriched uranium, plutonium, noble metals, and other components of the waste stream should be viewed as waste or potentially valuable resources. Several commentors asked that the EIS clarify which specific waste forms would be generated by the treatment processes. Others said the EIS should clarify whether the waste would remain at the Savannah River Site (SRS) after processing or be returned to Idaho if the PUREX process were used. Some commentors argued that the electrometallurgical treatment alternatives would not reduce the volume of waste to be stored in a repository. A few questioned how DOE can ensure the waste will meet the acceptance criteria for a repository when no one knows what those criteria will be—or if there will be any repository at all. A few others recommended that the EIS evaluate the PUREX process before it is shut down to ensure that the waste forms resulting from electrometallurgical treatment are as good as the borosilicate glass that is being prepared for the geologic repository.

Regarding the alternative technologies being evaluated as part of this EIS, the commentors generally agreed that DOE should evaluate in detail all of the alternative technologies that potentially could meet DOE's treatment and management needs—even those that DOE considers less technologically mature. Several commentors expressed the opinion that DOE already has made a technology decision in favor of electrometallurgical treatment, but that other alternative new technologies should not be dismissed because of a lack of knowledge about them. Some asked that the EIS: (1) explain how DOE can consider the PUREX process a reasonable alternative when, historically, it could not handle sodium-bonded spent nuclear fuel, and (2) evaluate whether changes in the PUREX process would be needed to accommodate sodium-bonded spent nuclear fuel. A few commentors suggested the EIS should analyze blanket and driver fuel separately, since they have different chemical and radiological characteristics and different treatments might be warranted.

Comments concerning environment, safety, and health issues were comparatively few, as were comments about transportation safety and security. A spokesman for the Shoshone-Bannock Tribe, which considers the Idaho National Engineering and Environmental Laboratory (INEEL) land to be part of their original territory, expressed confidence that the proposed electrometallurgical treatment process would not impact the land's cultural resources or native species. Other commentors wanted the EIS to explain whether there were any environmental threats associated with continued storage of the spent nuclear fuel in Idaho and the nature of the environmental impacts of all the alternative technologies listed in the Notice of Intent. Transportation-related comments were rare, but reflected some public concern about the safety and security of transporting spent nuclear fuel and other waste products over long distances.

Some commentors simply opposed the proposed action as a waste of money or an example of corporate welfare. Others stated that DOE already has determined its choice of alternatives and is merely engaging in a show process that meets the bare minimum legal requirements.

A.1.3 Comment Disposition and Issue Identification

Comments received during the scoping period were systematically reviewed and evaluated to determine whether the issues raised fell within or outside the scope of the EIS as contemplated in the Notice of Intent (64 FR 8553). Where possible, comments on similar or related topics were grouped under comment categories as a means of summarizing the comments. An attempt was made to avoid duplication in counting the number of comments received; however, comments submitted in both written and verbal form may have been counted

twice in some cases. The comment categories were used to identify specific issues of public concern. After the issues were identified, they were evaluated to determine whether they fell within or outside the scope of the EIS. Some issues were found to be already "in scope," i.e., they were among the EIS issues already identified by DOE for inclusion in the EIS. **Table A–1** lists these issues along with references to the specific EIS sections where each issue is discussed.

Additional issues were added to the scope of the EIS as a result of the public scoping process. These issues are listed in **Table A-2.**

DOE responded to all issues raised during the scoping period. Many of the public issues were not analyzed for a specific reason or were determined to be outside the scope of the EIS. These issues are listed in **Table A–3**. Corresponding responses from DOE also are provided in Table A–3 to explain why each issue was not analyzed.

Table A–1 Issues Already Included in the EIS (In Scope)

Issues	*No. of Comments*	*Draft EIS References*
The EIS should specify what the stable sodium compound technology alternative is and how it is derived	1	Section 2.3
The EIS should explain how the PUREX process, which could not handle sodium-bonded spent nuclear fuel before [in the aluminum-bonded Spent Nuclear Fuel EIS], now is considered an acceptable alternative for the proposed action.	1	Section 2.3.2
DOE says the Savannah River PUREX process will handle the sodium, but more research will be needed to improve the sodium-handling ability of the PUREX process. If research is needed to make the Savannah River PUREX process work for sodium, DOE might as well do research in Idaho in some different process. I'm in favor of Idaho; DOE should be cautious about talking PUREX and sodium-bonded stuff.	2	Section 2.3.2
The EIS should evaluate whether changes in the PUREX process would be needed to accommodate this material. After the plutonium is separated in the PUREX process, the high-level radioactive waste will be essentially no different from what is being handled now—no new ground broken, no new qualifications in materials. The uranium also will be unchanged after it goes through the PUREX process. The same with plutonium; if it goes through the PUREX, you haven't changed the existing process. So people should not get excited about this new stuff coming in—we've handled it for fifty years.	2	Sections 2.3.2 and 2.5.4
The EIS should analyze blanket and driver fuel separately since they have different chemical and radiological characteristics and different treatments might be warranted for each.	6	Sections 2.5, 4.3, 4.4, 4.5, 4.6, 4.7, and 4.8.
We're glad to see the melt and dilute alternative, a nonseparation technology, is being considered in this EIS.	1	Sections 2.5.5, 4.6, 4.7, and 4.8
The EIS should not assume that everything is known about the C-22 canister's performance in all conditions that could affect disposal; therefore, this canister should not be the only type of containment considered for encapsulation.	1	Section 4.13
The EIS should clarify whether, if the PUREX process were used, the waste would remain at the Savannah River Site after processing or be returned to Idaho.	4	Section 4.5.6
The EIS must clarify whether DOE considers low-enriched uranium to be a waste.	1	Section 4.3
The EIS must clarify which specific waste form will be used before any spent nuclear fuel is treated.	2	Sections 4.2.6, 4.3.6, 4.4.6, 4.5.6, 4.7.6, and 4.8.6
Will all of the technology alternatives shown on the poster handout be evaluated in this EIS? Has DOE made the ultimate decision concerning which alternatives will be evaluated in this EIS?	1	Section 2.5
Is there anything different about handling the materials involved in this EIS that would make the chloride volatility alternative more viable than was found for aluminum enriched uranium fuel? Hasn't this alternative already been evaluated in another EIS?	1	Section 2.7
The chemistry of the electrometallurgical process and the other alternatives should be provided.	1	Appendix C

Issues	*No. of Comments*	*Draft EIS References*
Blanket fuel can be mechanically declad and stripped of elemental sodium without the need for dissolution and separation of the solid fuel. While the minimal discussion in DOE documents stresses the difficulties of this approach, it is extremely hard to believe that the difficulties, costs, and risks of such minimal processing would be greater than those incurred by electrometallurgical treatment of the fuel. It is difficult to understand DOE's argument that this option is not as mature as electrometallurgical treatment, since it was employed for 15 times as many blanket rods as those that ultimately will be processed during the electrometallurgical treatment demonstration.	1	Section 2.5.3
Both DOE and the U.S. Nuclear Regulatory Commission underplay the significance of the mechanical decladding of 17 metric tons of heavy metal of blanket fuel. The U.S. Nuclear Regulatory Commission refers to this as a small amount even though it is 75 percent of the existing Experimental Breeder Reactor-II (EBR-II) blanket inventory. This is only one example of the loaded language in the Notice of Intent and its reference documents that strongly suggests the mechanical decladding alternative is not being fairly evaluated.	1	Section 2.5.3
All alternatives investigated and considered in this EIS should be viable and demonstrable. Unproven technologies preclude realistic bounding of environmental impacts and consequently do not appear to meet the intent of NEPA by providing implementable alternatives.	1	Section 2.5
Coordinate development of this EIS with others that are currently in preparation, including the Idaho High-Level Waste and Facilities Disposition, the Savannah River Spent Fuel, and the Yucca Mountain EISs.	3	Section 1.6
What are the plans for treatment of sodium-based fuel located at the other sites (about 2 percent of inventory)?	1	Section 2.2
Political decisions, such as the Idaho Settlement Agreement (which says that spent nuclear fuel must be out of Idaho by 2035), should not preclude any of the No Action Alternatives from being considered.	1	Sections 2.5.1, 4.2, and 4.13
I was pleased to hear you say you were looking at several options connected to the No Action [alternative].	1	Sections 2.5.1 and 4.2
The EIS should be specific about the stable compound of sodium and how that makes it like table salt (i.e., not a problem).	1	Appendix C and Section 2.3
How does this EIS relate to other EISs for treatment and disposal of other spent nuclear fuel types?	1	Section 1.6
What is the enrichment of the uranium?	1	Section 2.2.1
DOE should consider whether adequate information exists to allow estimation of bounding impacts for at least one treatment alternative in addition to the PUREX process at the Savannah River Site, the proposed electrometallurgical treatment at ANL-W, and the No Action Alternative. Instead of dismissing various treatment alternatives from further analysis, DOE should use existing information about those alternatives to support evaluation of as many treatment alternatives as possible. For example, the processing experience at Idaho Nuclear Technology and Engineering Center (INTEC) of the driver fuel using the PUREX-type process might be used in the analysis of the PUREX process at Savannah River.	1	Sections 2.5.3, 2.5.5, 4.4, 4.6, 4.7, and 4.8

Issues	No. of Comments	Draft EIS References
To support public review of the alternatives under consideration, the EIS should offer complete descriptions of how each alternative would be implemented.	1	Appendix C and Section 2.3
Each alternative should include full descriptions of all materials (including waste) resulting from treatment; proposed handling of all materials used in the treatment process; environmental impacts; measures to provide environmental protection; measures to ensure worker and public safety; facilities needed; full and complete discussion of waste handling facilities, magnitude and characteristics of the waste streams, type and amount of storage, and ultimate disposal method and location.	1	Sections 4.2, 4.3, 4.4, 4.5, 4.6, 4.7, and 4.8
The EIS should provide bounding estimates of the size, frequency, and number of expected shipments of products leaving Idaho on an annual basis.	1	Section 4.11
The EIS should provide bounding estimates of the duration of time that INEEL would store any products before shipment elsewhere after treatment.	1	Sections 4.2.6, 4.3.6, 4.6, 5.6, 7.1, and 8.0
Preparation of the EIS and the related decision-making process should be coordinated with related environmental documentation being prepared to ensure they are based on common data and common planning assumptions.	1	Section 1.6
The EIS should deal with disposition of all the waste streams resulting from this proposed action.	2	Sections 2.8, 4.2.6, 4.3.6, 4.4.6, 4.5.6, 4.6.6, 4.7.6, and 4.8.6
To help the public understand DOE's rationale for moving forward with this decision, the EIS should describe how each treatment alternative would address the waste acceptance criteria for resulting waste products destined for disposal at current and planned disposal facilities.	1	Sections 2.8 and 4.13
The draft EIS should include a complete subject index and not just an alphabetically arranged list of headings.	1	Chapter 9
DOE should coordinate the related projects [e.g., the Idaho High-Level and Facilities EIS; the Management of Savannah River Spent Nuclear Fuel EIS; and the Geological Disposal Repository for Spent Nuclear Fuel and High-Level Waste at Yucca Mountain, Nevada, EIS] to support consistent, coordinated decision-making.	1	Section 1.6

Table A–2 Issues Added to the Scope of the EIS

Issues	*No. of Comments*	*Draft EIS References*
Analyses related to the No Action Alternative should include the environmental consequences of not doing anything...and [this alternative] should not be written off because somebody made a political decision that this stuff will be out of Idaho by 2035.	1	Section 4.2
The proposed structure of the EIS as described in the Notice of Intent is inconsistent with DOE's approach to spent nuclear fuel management at other sites and prematurely promotes a preferred option for managing sodium-bonded spent nuclear fuel. By presuming the proposed action is electrometallurgical treatment, the proposed structure of the EIS effectively establishes this treatment as the preferred alternative for stabilization of this material. While it is reasonable to rule out obviously impractical alternatives in the scoping process, several of the alternatives described in the Notice of Intent are technically viable and should not be prematurely dismissed.	3	Sections 1.2, 1.3, 1.4, and 2.5
DOE should consider the possibility of using different treatment processes for treatment of the driver fuel and the blanket fuel. Could the driver fuel be handled as part of the ongoing demonstration? Treatment alternatives for the blanket fuel could conceivably include direct disposal, as it is not yet clear that it will require treatment before disposal.	1	Sections 2.5.3, 2.5.4, 2.5.5, and 2.5.6
The three alternatives presented for treatment of the EBR-II fuel are the most reasonable ones politically available, namely (1) separate the highly enriched uranium and make the other materials into a ceramic using a hot isostatic press, or (2) separate both the uranium and plutonium using the PUREX process at the Savannah River Site and...vitrify the waste, or (3) direct burial.	1	Sections 2.5, 4.2, 4.3, and 4.4

Table A–3 Other Issues Considered

Issues	*No. of Comments*	*DOE Responses*
Costs		
The public needs information about the cost of the proposed action and the costs of the other technology alternatives before it can adequately comment on the EIS.	6	Information on cost will be made available to the public via the Cost Study, which will be issued during the draft EIS public comment period.
This program is not worth the money it will cost.	1	Information on cost can be found in the Cost Study which, along with the EIS, will factor into the Record of Decision.
The cost assessment has to be part of the EIS.	2	Although the cost assessment is not part of the EIS, it has been prepared concurrently with the EIS. The Cost Study, along with the EIS, will factor into the Record of Decision.

Issues	*No. of Comments*	*DOE Responses*
If you don't account for the low-enriched uranium stream, your cost estimates are going to be wrong or at least off. If you don't have a disposition scenario, you have to look at the long-term economic and environmental storage costs that will belong to DOE for a long time.	2	The environmental impacts and cost of storage of the low-enriched uranium stream have been analyzed in the EIS and Cost Study, respectively.
We think that combining the research and development efforts on these two different types of fuel [blanket and driver] might lead to considerable cost savings.	1	If an alternative technology is chosen that could treat both the driver and blanket fuel, research and development efforts would be combined, as they were for electrometallurgical treatment research and development.
As Savannah River has a huge vitrification facility and that technology already is available, DOE should compare the costs of vitrification with the costs of the PUREX process.	1	The vitrification facility at SRS treats the high-level radioactive waste that results from PUREX processing. The two are not independent. The cost of vitrification will be included in the cost of the PUREX alternative in the Cost Study. Direct vitrification of sodium-bonded spent nuclear fuel, however, is not technically feasible.
Cost analysis should include: (1) program costs so far in detail, including whether these costs were for pyroprocessing or for the EBR-II to shut down; (2) how much it would cost to close out the program at the end of the test, including decommissioning the machinery and dealing with all the waste streams (such as low enriched uranium); (3) what it would cost to scale-up the program, including commissioning and dealing with all waste streams at the end of the scale-up.	1	The Cost Study does not include EBR-II shutdown costs. The Cost Study includes the cost of any new machinery, if needed; treating the sodium-bonded spent nuclear fuel; deactivating machinery; and dealing with the waste streams. The low-enriched uranium product is not a waste. Its disposition will be the subject of a future NEPA review, however, the cost of storage of the low-enriched uranium is included in the Cost Study.
The EIS should include the cost of transportation if this stuff is moved across country from Idaho to South Carolina and then from South Carolina to wherever.	1	The cost of offsite and onsite transportation is included in the Cost Study.
Environment, Safety, and Health		
The Shoshone-Bannock Tribe considers the INEEL land to be part of their original territory and believes the electrometallurgical treatment process will not impact the land's cultural resources or native species and will make the best uses of these resources.	1	The commentor's support for the electrometallurgical technology is acknowledged.
DOE should explain the environmental considerations that are pushing this EIS to completion in such a short period of time, including the environmental threats of continuing to store the EBR-II spent nuclear fuel in Idaho, if any. Then, DOE should compare these environmental threats with the R&D schedule for all the alternative technologies being considered, especially the nonseparation technologies.	1	The purpose and need for agency action is discussed in Section 1.2. Under the No Action Alternative, the Department may decide to continue to store the sodium-bonded spent nuclear fuel indefinitely, or until research and development of an alternative treatment technology is successfully completed.

Issues	*No. of Comments*	*DOE Responses*
DOE should be able to provide the environmental impacts for all of the alternative technologies listed in the Notice of Intent; they should not be dismissed because DOE does not know enough about them.	1	Alternative technologies were not dismissed solely based on the lack of available information on the respective technologies. As discussed on Section 2.6, chloride volatility was dismissed due to the potentially significant (in comparison to other treatment technologies) occupational and public risks from the volatilization of fission products and chloride gas.
Nonproliferation		
Nonproliferation should not be addressed in a separate report; the nonproliferation assessment should be part of the EIS. Short-circuiting the nonproliferation analysis is particularly egregious in light of the pledge in the Notice of Intent to include this assessment in the draft EIS and the existence of such a DOE assessment from December 1998.	3	The Notice of Intent stated, "The combination of the information contained in the draft EIS, the public comment in response to the draft EIS, and the Nonproliferation Impacts Assessment report will enable the Department to make a sound decision..." Although the Nonproliferation Impacts Assessment is separate from the EIS, it will fully analyze the nonproliferation impacts of the alternatives in the EIS.
The public should have an opportunity to comment on the ongoing nonproliferation assessment, and the assessment should be publicly available before the comment period is closed on this EIS.	9	The Nonproliferation Impacts Assessment will be available to the public prior to the end of the public comment period for this draft EIS. However, the assessment will be issued as a final document.
The public needs information about the nonproliferation impacts of the proposed action before it can comment on the EIS.	1	The Nonproliferation Impacts Assessment will be available to the public prior to the end of the comment period for this draft EIS.
The EIS should not be released until nonproliferation concerns no longer are being debated; there is a potential for exporting this technology.	1	The Nonproliferation Impacts Assessment will be available to the public prior to the end of the comment period for this draft EIS.
Given that obtaining fuel material is the greatest hurdle to producing nuclear weapons, DOE should take nonproliferation concerns about small-scale reprocessing technologies like pyroprocessing more seriously and give them greater weight in its decision-making.	2	DOE is concerned with the nonproliferation impacts of all of its proposed actions. It is for this reason that a separate Nonproliferation Impacts Assessment report will be prepared specifically to address the alternatives under consideration.
Pyroprocessing is a proliferation-prone technology. For example, although plutonium no longer would be separated as a separate step in the EBR-II treatment, the original pyroprocessing technology was intended to remove plutonium and actinide components in a liquid cadmium cathode, and that option is always there.	4	DOE has conducted four independent nonproliferation assessments of electrometallurgical technology over the past 11 years. A new assessment that addresses the alternatives under consideration for treating sodium-bonded spent nuclear fuel is being conducted concurrently with the EIS and the report will be available for public review. Previous assessments have concluded that electrometallurgical technology was not capable of separating plutonium in a form that would be suitable for weapons. Development of the liquid cadmium cathode was canceled before significant engineering issues were resolved. No liquid-cadmium cathode was ever completed for the electrorefiners used in the Fuel Conditioning Facility, where spent nuclear fuel treatment would take place.

Issues	No. of Comments	DOE Responses
Pyroprocessing will continue to search for other missions before the issue of whether it can be shut down and decommissioned on a timely basis is decided. Use of pyroprocessing should be "nipped in the bud" because of nonproliferation concerns.	1	Electrometallurgical treatment technology is a promising technology for the management of spent nuclear fuel. DOE is considering applying this technology for the management of some or all of its sodium-bonded spent nuclear fuel at sometime in the near future. DOE is conducting a Nonproliferation Impacts Assessment that focuses on the application of electrometallurgical and alternative treatment technologies to sodium-bonded spent nuclear fuel. This new assessment will be made available to the public during the draft EIS public comment period. Previous nonproliferation assessments have found electrometallurgical technology to be in accordance with the U.S. nuclear nonproliferation policy for the specific applications considered.
The Savannah River nonproliferation assessment states that pyroprocessing can be modified to produce plutonium. This modification may not be easy, but it would be easier than building an entire PUREX facility or adding such a capability to any of the other nonseparation technology options—and it would certainly be of interest to rogue states who are interested in producing nuclear weapons.	3	The modification referred to in the Savannah River nonproliferation assessment involves adding a proven aqueous process such as PUREX onto the electrometallurgical process. Because the aqueous processes would be incompatible with the dry inert atmosphere required by the electrometallurgical process, a separate facility would be required. If a nation bent on weapons production had this capability, it could separate weapons-usable plutonium directly from spent nuclear fuel or plutonium production targets without the need for the electrometallurgical process equipment.
This program is inconsistent with the present U.S. position on reprocessing. The United States should not be funding new separation technologies.	2	The DOE Office of Arms Control and Nonproliferation will assess the nonproliferation impacts of the alternative treatment technologies under consideration in this EIS in a separate report to determine if the alternatives are consistent with U.S. nonproliferation policy and goals.
Pyroprocessing is reprocessing. MacArthur Prize Fellowship winner Frank Von Hippel and Professor James Warf, inventor of several reprocessing technologies, underscore this fact and express concern about the nuclear nonproliferation impacts of pyroprocessing: "...because pyroprocessing facilities are more compact than conventional facilities, they are easier to conceal. The world would become a more dangerous place."	2	In a nonproliferation assessment conducted for DOE in 1992, a panel of experts stated that there was no reason to conclude that electrometallurgical process facilities would be any easier to conceal than a conventional reprocessing plant. The electrometallurgical process requires a large heavily shielded hot cell with highly purified argon atmosphere and specialized process equipment.
While the Notice of Intent states that DOE has no plans to apply this technology (electrometallurgical treatment) to any other types of spent nuclear fuel, it clearly leaves the door open for other applications and raises the concern that ANL-W will continue to hunt for other materials that can be used to keep the electrometallurgical treatment apparatus operating after the sodium-bonded fuel campaigns are completed, or even to justify construction of new facilities. This open-ended approach...has severe implications for nonproliferation.	1	Electrometallurgical treatment technology is a promising technology for the management of spent nuclear fuel. DOE is considering applying this technology for the management of some or all of its sodium-bonded spent nuclear fuel at sometime in the near future. DOE is conducting a Nonproliferation Impacts Assessment that addresses the application of electrometallurgical technology, as well as the other alternatives under consideration, to sodium-bonded spent nuclear fuel. This new assessment will be made available to the public during the draft EIS comment period. Previous nonproliferation assessments have found electrometallurgical technology to be in accordance with U.S. nuclear nonproliferation policy for the specific applications considered.

Issues	*No. of Comments*	*DOE Responses*
The electrometallurgical treatment process can be modified to produce plutonium. Moreover, there are no plans to place ANL-W facilities under international safeguards. Therefore, from an arms control standpoint, the Fuel Conditioning Facility must be regarded as a dual-use facility capable of being operated as a reprocessing plant. In view of this, it is highly advisable to prepare for timely shutdown of the facility when any campaigns for which it is determined to be essential (if any) are completed.	1	DOE has conducted four independent nonproliferation assessments of electrometallurgical technology. A new assessment that focuses on the application of electrometallurgical technology to sodium-bonded spent nuclear fuel is being conducted concurrently with the EIS and will be available for public review. Previous assessments have concluded that electrometallurgical technology was not capable of separation plutonium in a form that would be suitable for weapons. Development of the liquid cadmium cathode was canceled before significant engineering issues were resolved. No liquid-cadmium cathode was ever completed for the electrorefiners used in the Fuel Conditioning Facility, where the spent nuclear fuel treatment would take place. The Fuel Conditioning Facility operates under DOE safeguards and security requirements.
DOE should make the nonproliferation assessment of the proposed electrometallurgical treatment action a part of the NEPA process. The assessment should cover not only the proposed action, but the broader proliferation implications of continued research and development of this reprocessing technology.	1	DOE is concerned with the nonproliferation impacts of all of its proposed actions. It is for this reason that a separate Nonproliferation Impacts Assessment will be prepared that will specifically address electrometallurgical treatment technology. DOE will consider this assessment in its decision-making process.
One issue that should be covered in the nonproliferation assessment is whether promotion of electrometallurgical treatment as a "proliferation-resistant" technology ultimately will prove harmful to U.S. nonproliferation goals. If this designation does not have a sound technical basis (as we believe it does not), the ultimate result will be an increased danger of proliferation.	1	DOE is concerned with the nonproliferation impacts of all of its proposed actions. It is for this reason that a separate Nonproliferation Impacts Assessment will be prepared that will specifically address electrometallurgical treatment technology.
For nations that reprocess spent nuclear fuel, switching to electrometallurgical treatment may enable them to argue that their current safeguards burden should be relaxed.	1	Prior to the export of any technology that may have nonproliferation impacts to a foreign nation, DOE assesses the impacts, if any, to ensure that U.S. nonproliferation goals are met.
The EIS should include a detailed, thorough analysis of the weapons proliferation implications of each treatment alternative.	1	DOE's Office of Arms Control and Nonproliferation is preparing a Nonproliferation Impacts Assessment of each treatment alternative. This new assessment will be made available to the public during the draft EIS public comment period.
One of the justifications for proceeding with the mixed oxide (MOX) proposal was to satisfy the international community's desire to forestall the ready availability of weapons-grade materials. This proposal creates the ready availability of those same materials. The EIS must account for this apparent contradiction of policy and address the measures intended to safeguard the by-product(s) of this process.	1	DOE recognizes the need to identify nonproliferation impacts of the treatment technologies. Therefore, the DOE Office of Arms Control and Nonproliferation will assess the nonproliferation impacts of the alternative treatment technologies in a report, separate from this EIS.
Alternative Technologies		
The EIS should re-evaluate and address plutonium separation; it would be less expensive to separate the plutonium because that would mean the repository would need to last only 300 years, instead of 10,000.	1	The EIS is evaluating plutonium separation as a part of the PUREX option for the blanket fuel. Plutonium separation would not guarantee a different performance requirement for the repository, since the long-term requirements are driven by other radioisotopes.

Issues	*No. of Comments*	*DOE Responses*
DOE has already made up its mind. Other methods than pyroprocessing haven't been given sufficient attention. These alternative methods continually are slated as "not developed enough." Yet in three years, there hasn't been much attention given to developing them to a point where they could be reviewed fairly. Alternative new technologies should not be dismissed due to lack of knowledge about them.	4	In response to public comments, DOE has reformulated the scope of the EIS to address more generally the treatment and management of DOE sodium-bonded spent nuclear fuel. Information developed in the course of preparing this EIS suggests that alternative technologies may have certain advantages (e.g., cost) for some or all of the fuel. Accordingly, DOE did not identify a preferred alternative in the Draft EIS. In the EIS, DOE also considers an option under the No Action Alternative in which the Department would actively conduct research and development of promising new technologies.
The Notice of Intent is biased toward electrometallurgical treatment because it disparages the other alternatives, which are tacked on just to satisfy a legal requirement. The program is taking the wrong approach toward electrometallurgical treatment because the alternatives are not really valid.	2	In response to public comments, DOE has reformulated the scope of the EIS to address more generally the treatment and management of DOE sodium-bonded spent nuclear fuel. Information developed in the course of preparing this EIS suggests that alternative technologies may have certain advantages (e.g., cost) for some or all of the fuel. Accordingly, DOE did not identify a preferred alternative in the Draft EIS. In the EIS, DOE also considers an option under the No Action Alternative in which the Department would actively conduct research and development of promising new technologies.
There is a danger that other technologies will be abandoned if, as it appears, DOE is rushing to produce waste or materials to go to a waste site somewhere or is pushing pyroprocessing ahead of other technologies.	1	In response to public comment, DOE has restructured the alternatives to be considered, including an option of deferring a treatment decision and developing alternative technologies.
The EIS should identify the alternative sites if Idaho is not selected and which sites will be needed for the alternative technologies.	1	The EIS has identified the SRS as an alternative site for the PUREX and melt and dilute alternatives.
The EIS should include a stabilization timeline on environmental grounds for EBR-II spent nuclear fuel. The time line should include the time needed to more fully develop other alternatives.	2	EBR-II spent nuclear fuel must be removed from the State of Idaho by the year 2035 in accordance with a DOE/State of Idaho Settlement Agreement and Consent Order, signed in October of 1995. DOE believes that treatment to remove sodium from EBR-II and other spent nuclear fuel will make acceptance of this fuel in a national geologic repository much more likely.
Will the EIS look at the vitrification facility at INTEC?	1	The proposed Vitrification Facility at INTEC is not compatible with any of the proposed waste forms or metal fuel such as the EBR-II or Fermi-1 fuel. It is for this reason that DOE has not analyzed this facility in the EIS.
The EIS should address the size of the electrometallurgical treatment facility and whether the plant capacity is greater than needed for the proposed mission (more than 62 metric tons of heavy metal).	1	The plant capacity for treating spent nuclear fuel using the electrometallurgical treatment equipment is approximately 5 metric tons of heavy metal per year. It would therefore require 12 years to treat the entire 60-metric ton DOE sodium-bonded spent nuclear fuel inventory.
The Notice of Intent indicates that DOE has no plans to apply electrometallurgical treatment to any other spent nuclear fuel types, suggesting the plant would be decommissioned after completing the electrometallurgical treatment mission for sodium-bonded spent nuclear fuel. The EIS, therefore, should address the impacts of decommissioning the plant.	2	At this time, DOE has no intent to apply electrometallurgical treatment to any other spent nuclear fuel types. The electrometallurgical treatment process equipment is housed within a large multipurpose hot cell facility which has programmatic value to DOE, even in the absence of a spent nuclear fuel treatment program. Any specific electrometallurgical treatment equipment would be deactivated at the end of any treatment program; however, there are no plans to discontinue use of the hot cell facility.

Issues	*No. of Comments*	*DOE Responses*
Use a reactor or accelerator to fission the transuranic material.	1	This is not a reasonable alternative because the transuranic materials resulting from the electrometallurgical treatment process would require extensive additional processing before they would be suitable for fission in a reactor.
Adding another furnace and cathode to ANL-W's facility would both accelerate the processing and provide opportunities for new research.	1	The existing electrometallurgical treatment equipment would provide DOE an adequate processing rate for the sodium-bonded spent nuclear fuel inventory. New research would be accomplished with equipment in a nonradioactive laboratory environment.
Regarding the use of melt and dilute and Savannah River—the Savannah River process will not be sized or configured to handle INEEL fuel (which should be contrary to the Foreign Research Reactor Record of Decision). Melt and dilute at INEEL solely should be the alternative.	1	The sodium-bonded fuel would have its cladding and sodium removed before being placed in aluminum cans for shipment to the SRS, where the proposed melt and dilute process would take place. This pretreatment step would make the fuel compatible with the proposed SRS process.
Sodium is highly reactive with water/moisture, and this property could be taken advantage of by controlled reaction on a limited scale—exposing the sodium-bonded material to moisture. The sodium hydroxide formed could be neutralized with an appropriate acid, allowing the remaining spent nuclear fuel to loose its pyrophoric properties. Please address this in the EIS.	1	For fuel in which the sodium can be exposed, the EIS describes a process for safely removing it by vacuum distillation. The process described in the comment would accelerate corrosion of the uranium, resulting in an unsafe pyrophoric condition.
DOE may want to consider an alternative that examines the relationship between the EBR-II fuel at INEEL and the high-level radioactive waste at the stabilization facility.	1	The proposed INEEL high-level radioactive waste management EIS is considering methods to manage the calcine that was produced from the reprocessing of DOE spent nuclear fuel at INTEC. With the decision to shut down the reprocessing facilities, no processes are currently available that would make the sodium-bonded fuel compatible with the calcine.
The fall 1996 National Research Council report on pyroprocessing at ANL states that even more time and money than originally planned will be needed to "achieve the program's objectives" and raises troubling questions about several aspects of the research itself. Later reports, unfortunately, do not specifically follow up on these concerns.	1	DOE's Electrometallurgical Research and Demonstration Project has addressed concerns that have been raised by the National Research Council. Their 1998 report has recognized the progress in the Demonstration and has stated it should continue to completion.
The fall 1996 National Research Council report raises serious concerns about several aspects of the research including a lack of coordination between ANL East and West. This lack of coordination and differing goals have led to duplicate efforts in at least one case and equipment failures. The report notes the lack of a "well-coordinated implementation plan between ANL East and West...."	1	DOE's Electrometallurgical Research and Demonstration Project has addressed concerns that have been raised by the National Research Council. Their 1998 report has recognized the progress in the Demonstration and has stated it should continue to completion. DOE's Electrometallurgical Research and Demonstration Project, which is nearing completion at ANL-W, has successfully met National Research Council criteria to date. The success of this demonstration project has been possible only through close coordination between scientists and engineers at ANL-East and -West.

Issues	*No. of Comments*	*DOE Responses*
The [fall 1996 National Research Council] report found that equipment is not performing at expected levels and separation efficiencies are lower than expected. This means that, so far, the basic goal of the pyroprocessing program—to separate the uranium from the rest of the irradiated fuel—has not been met.	1	DOE's Electrometallurgical Research and Demonstration Project has addressed concerns that have been raised by the National Research Council. Their 1998 report has recognized the progress in the Demonstration and has stated it should continue to completion.
Research on selected alternatives should have been carried out to support a defensible analysis of their feasibility in the EIS.	1	The alternatives to be analyzed in detail are described in Chapter 2 of the EIS. An analysis of their feasibility is included in this chapter.
DOE has not demonstrated there is a safety-based need to process the driver fuel by experimentally assessing the impact of elemental sodium on radionuclide leach rates.	1	DOE has proposed treatment to remove the sodium from sodium-bonded spent nuclear fuel to allow acceptance of this fuel in a national geologic repository. This is because sodium reacts with water in the environment to form corrosive sodium hydroxide solutions and potentially explosive hydrogen gas.
DOE should initiate a process similar to the Processing Needs Assessment to determine at the earliest possible date the "small quantities of certain spent nuclear fuel types" that may be considered for electrometallurgical treatment in the future. Such an effort is essential for shutdown and decommissioning planning.	1	At this time DOE has no intent to apply electrometallurgical treatment to any other spent nuclear fuel types. If, during the sodium-bonded fuel treatment program, DOE finds another application for electrometallurgical treatment at ANL-W, the development of plans to deactivate the electrometallurgical treatment equipment at ANL-W would be delayed accordingly.
A study similar to the 1997-98 Processing Needs Assessment should be conducted to identify all materials in the DOE complex that might need reprocessing in the Savannah River Site canyons for stabilization purposes, thus limiting the universe of potential uses for the canyons and facilitating planning for their shutdown. A similar process should be conducted for the Fuel Conditioning Facility as part of this EIS process, with the opportunity for full public participation and comment.	1	The EIS is being coordinated with other DOE EIS documents and Records of Decision concerning complex-wide management of spent nuclear fuel. These EISs are described in Section 1.6 of this EIS.
It is unfortunate that the option of separating the plutonium along with the uranium by the electrometallurgical process could not have been considered. Although the resulting fissile material would only have been suitable for a fast-neutron reactor...at least we would not have the agony of worrying about putting this plutonium in a repository.	1	The electrometallurgical process cannot separate plutonium. Because of potential nonproliferation implications, the Department elected not to develop the capability for electrometallurgical processing to produce any plutonium-bearing product. Plutonium separation is an integral part of Alternative 3, PUREX Processing of the Blanket Fuel at SRS. However, removal of the plutonium would not significantly affect the long-term performance of the repository, which is driven by other radioisotopes.
Since the electrometallurgical method works, is ready to go, and is not expensive, it is in the public interest to get the fuel treatment job done rather than delay while developing some other method.	1	The commentor's support of the electrometallurgical treatment technology is acknowledged.

Issues	*No. of Comments*	*DOE Responses*
The addition of depleted uranium to the electrometallurgical treatment process is both a waste of depleted uranium and enriched uranium. Why add the depleted uranium?	1	Blending depleted uranium with the highly enriched uranium recovered from the spent EBR-II driver fuel results in low-enriched uranium. This step, which is consistent with U.S. nonproliferation policy, results in lower costs for storing and safeguarding the uranium. Because the uranium ingots still contain more enrichment than is required for commercial power reactor fuel, their potential economic value is not decreased. The Department currently stores more than 500,000 tons of depleted uranium for which no immediate use is planned. Using some 10 tons of this inventory for treating spent nuclear fuel would have no discernable impact.
Waste		
The EIS should address the disposal specifications for spent nuclear fuel, and DOE should make sure that, whatever technology is selected, the spent nuclear fuel will meet repository specifications. This determination should be made before the canyons are shut down to avoid precluding a way to get rid of the materials.	1	The ceramic and metallic high-level radioactive waste forms that would be produced from the proposed action are expected to be at least as durable as the borosilicate glass high-level radioactive waste form. The design criteria for the national spent nuclear fuel repository include receipt and disposal of the borosilicate glass high-level radioactive waste.
The EIS should explain why stainless steel and noble metals are considered waste and not potentially valuable resources.	1	The stainless steel and noble metals would be part of the metallic high-level radioactive waste forms. High-level radioactive waste is a material that the U.S. Nuclear Regulatory Commission has determined requires permanent isolation.
Waste characterization is a problem. Low enriched uranium is a problem-it's a waste not a product. The EIS should look at the long-term storage costs of uranium.	2	DOE does not consider low-enriched uranium to be a waste. No highly enriched uranium would result from any of the alternatives considered at INEEL.
Discussion of the low-enriched uranium stream must include a full analysis of what happens to this stream and when.	1	DOE has not made a decision concerning future uses for the low-enriched uranium other than that the low-enriched uranium would not be used for defense purposes.
Spent nuclear fuel is not a waste.	1	Spent nuclear fuel is a fuel that has been withdrawn from a nuclear reactor following irradiation; the constituent elements have not been separated for reprocessing.
The project is being sold as a way to reduce the volume of waste to Yucca Mountain. It won't reduce actual volume; it will only increase floor space by putting ceramic and metallic waste forms closer together while still avoiding criticality issues. That's where your 65 percent comes from. You don't have volume reduction; you just have split the waste into lots of different forms which you still have to find a home for. But the message that is getting out is that you will be sending a smaller by weight number of packages to Nevada.	3	Waste volumes, masses, and disposal paths for all types of waste are considered for the different alternatives in this EIS. The volume of high-level radioactive waste or spent nuclear fuel that would be sent to a geologic repository are some of the things considered in the waste management sections. The potential impact on different disposal sites is considered and discussed. However, the purpose and need for the proposed action is to treat and manage the spent fuel, not to reduce the volume of waste that eventually will be sent to a repository.
DOE does not know if electrometallurgical treatment waste will meet the repository waste acceptance criteria. DOE does not know what those criteria will be—or if there will be any repository at all. Will the waste be acceptable? We need honest assumptions on the waste stream.	4	The repository waste acceptance criteria are still being developed. However, the ceramic and metallic waste forms that would result from the electrometallurgical treatment process are expected to be accepted into the repository.

Issues	No. of Comments	DOE Responses
DOE should consider dealing with this high-level radioactive waste as part of the high-level radioactive waste being dealt with at INTEC.	1	The proposed INEEL High-Level Radioactive Waste Management EIS is considering methods to manage the calcine that was produced from the reprocessing DOE spent nuclear fuel at INTEC. With the decision to shut down the reprocessing facilities, no processes are currently available that would make the sodium-bonded fuel compatible with calcine. The restart of these facilities was considered and eliminated from the alternatives.
DOE admits to having no knowledge of the whereabouts of the documents pertaining to previous removal of the sodium bonding from 17 metric tons of EBR-II blanket fuel via mechanical decladding. Such mismanagement, if true, is of concern and should be investigated. We request that a greater effort be undertaken to find these documents and make them publicly available during the EIS period.	1	DOE has found the documents that describe the process, equipment, operating procedures, and waste disposal paths for the decladding and sodium removal of the 17 metric tons of EBR-II blankets. These documents were considered during the selection of the proposed decladding and sodium removal alternatives.
DOE's plans for disposing of the low-enriched uranium created from this process—will it be stored as a waste or sold as a resource?	2	DOE has not made a decision concerning future uses for the low-enriched uranium produced by the electrometallurgical treatment other than the decision that the low-enriched uranium would not be used for defense purposes.
This program [electrometallurgical treatment] has no place in a sound nuclear waste management policy. Proponents of this program are . . . making the problem worse not better. This program will increase the complexity and amount of nuclear waste generated at ANL. We do not support an expansion of this program and urge that it be terminated.	1	DOE believes that treating sodium-bonded spent nuclear fuel is in keeping with sound nuclear waste management. This is because the proposed action would reduce uncertainty regarding waste disposal. Also, the number of canisters that must be disposed of in a geologic repository would be reduced. Further, ceramic and metallic waste material is very durable and has been formulated to be unreactive in the environment.
If DOE creates high-level radioactive waste in a vitrified form, there will be three forms of high-level radioactive waste in one Idaho county (ceramic, metal, vitrified).	2	The statement is correct. Different waste streams often require different stabilization techniques. The ceramic, metallic and vitrified waste forms are being developed because they are best suited for specific waste streams.
If this material won't meet the disposal specifications for the repository, a specification should be incorporated into the Record of Decision to say that DOE will look at this material and its proposed specifications before the canyons are shut down to ensure it is as good as the PUREX borosilicated glass that is being prepared for the Yucca Mountain repository.	1	DOE will consider the programmatic impacts including schedule and technical uncertainties such as availability and waste acceptance when a Record of Decision is made.
Since the waste acceptance criteria at Yucca Mountain currently is not confirmed, how do you intend to meet and store [the waste] for "road-ready" conditions?	1	The present goal is to place the spent nuclear fuel and high-level radioactive waste at ANL-W in retrievable storage so that it can be shipped to the proposed packaging facility that will ship the INEEL-DOE spent nuclear fuel to the repository. For the SRS alternatives, the high-level radioactive waste glass or melt and dilute product would be coordinated with the streams that will be produced at SRS.

Issues	*No. of Comments*	*DOE Responses*
Will planned dry storage have to be retreated later to meet acceptance criteria at Yucca Mountain?	1	The No Action Alternative may require future treatment. The goal of the other alternatives is to put the waste in road-ready condition without further treatment. The uncertainty in the final repository waste acceptance criteria is part of the programmatic considerations.
Uranium metal also is reactive; will it be treated before placement in a geologic depository?	1	Uranium metal is currently managed as part of the Materials Disposition program and is out of the scope of the EIS.
The Environmental Assessment contained ridiculous estimates of waste streams, especially the low-level radioactive waste streams. Actual information about waste generated from the demonstration project should be released to the public for use in the EIS.	1	The actual waste generation rates for the demonstration project have been used to calculate estimates of waste streams in this EIS.
Previous National Research Council reports have concluded that several of the waste forms generated by this technology [pyroprocessing] would not be suitable for placement in a geologic repository. The fall 1996 National Research Council report raises serious concerns about the testing procedures used to determine whether one of the new waste forms will be suitable for placement in a geologic repository. Most troubling of all is the analysis of ANL's choice of test protocol. A key issue is the release of the radionuclides from the waste. The report notes that the test protocol focuses on a radionuclide release mechanism that is... "incorrect at best, and potentially misleading at worst."	1	In order to address the question on waste form qualification, DOE has asked the National Research Council to conduct a specific review on this subject. The report that discusses the results of this waste qualification review and the other National Research Council reports will be considered when a record of decision is formulated.
Since getting waste ready for a geologic repository is the justification for this project, it must not go forward until the waste produced by the demonstration project has been fully characterized, which will occur early in the next century.	1	The uncertainty and status of each waste or spent nuclear fuel characterization are part of the programmatic consideration when a record of decision is formulated.
Spent nuclear fuel must be removed by 2035 as a result of processing. One concern is that transuranic waste will go to the repository, but low-enriched uranium and highly enriched uranium will stay at INEEL.	1	No highly enriched uranium would result from any of the alternatives considered at INEEL. DOE has not made a decision concerning future uses for the low-enriched uranium other than the decision that the low-enriched uranium would not be used for defense purposes. DOE will compare all reasonable alternatives on the basis of cost, including the cost of long-term storage of materials.
Compare heat loading with the ceramic and metallic waste forms to heat loading of the highly enriched uranium rods—are they comparable with commercial spent nuclear fuel?	1	As packaged for disposal in a geological repository, the heat loading for the ceramic and metallic waste forms is higher than that for the highly enriched uranium fuel because of fissile material limits for disposal packages. These high-level radioactive waste packages in general have lower heat loads than commercial spent nuclear fuel. Heat load would not be a concern regarding potential disposal in a geologic repository.
Transportation		
These materials should not be transported throughout the United States.	1	It is DOE's intention to minimize transport of radioactive materials associated with its sodium-bonded spent nuclear fuel inventory wherever possible.

Issues	*No. of Comments*	*DOE Responses*
If the ultimate burial place for the high-level radioactive waste is 1,000 miles away instead of 2,000 miles away, is that fact insignificant to transportation?	1	Generally, the environmental impacts of transporting spent nuclear fuel and high-level radioactive waste are small and would not differ significantly under the example posed by the commentor. DOE recommends the commentor see the *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement* for additional information on this subject.
The EIS should evaluate the potential for terrorism, especially during transportation.	2	The potential for terrorist acts involving material transports does not fall within the scope of this EIS.
Is it not known that, if the waste is sent to South Carolina [SRS], it will have to go somewhere else eventually; it won't stay in South Carolina?	1	As described in Section 2.5 of the EIS, Alternatives 3 and 5 would result in the storage of waste or by-products at SRS in South Carolina. For Alternative 3, the products from processing blanket fuel in the PUREX facility would be plutonium metal, borosilicate glass logs, and depleted uranium. For Alternative 5, the metallic waste product from the blanket fuel melt and dilute process would be stored in the L Area at the SRS.
The EIS should provide bounding estimates of the size, frequency, and number of expected shipments of products coming into Idaho.	1	Chapter 4 and Appendix G of the EIS provide estimates of the size, frequency, and number of expected shipments of products coming into Idaho. The Record of Decision for the 1995 *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement* also describes the size, frequency, and number of spent nuclear fuel shipments coming to Idaho.
DOE should develop an agreement with the Shoshone-Bannock Tribes to allow and appropriately manage the transport of any radioactive materials across the reservation.	1	Regardless of the alternative chosen, DOE will proceed in accordance with the DOE/Shoshone-Bannock Tribes Agreement-in-Principle, which covers notification and coordination of the transport of radioactive materials across the Fort Hall Reservation.
EIS Schedule		
This EIS may not be needed because the 1996 Environmental Assessment may be adequate.	1	DOE prepared an environmental assessment for the demonstration of electrometallurgical treatment on a limited amount(1.6 metric tons) of sodium-bonded spent nuclear fuel. In the May 15, 1996 Finding of No Significant Impact for the Environmental Assessment, DOE committed to prepare an EIS before applying the electrometallurgical treatment technology to the production-scale treatment of the sodium-bonded spent nuclear fuel inventory.
The Draft SBSNF EIS should not be issued for public comment before publication of relevant reports (e.g., waste qualification) from the National Research Council or the ongoing nonproliferation study. The schedule implies that DOE is not interested in incorporating the results from these studies into the EIS. Therefore, the time line for the EIS should delay its completion until at least three months after completion of these studies.	5	The Electrometallurgical Research and Demonstration Project is scheduled to conclude in August of 1999. At that time DOE will know if it has met the success criteria established by the National Research Council for the electrometallurgical treatment demonstration. Publication of the final report on the electrometallurgical treatment demonstration by the National Research Council may require a few months past the end of the demonstration project. DOE expects that the report will be available before it makes a decision on the management of the sodium-bonded spent nuclear fuel. DOE has prepared a Nonproliferation Impacts Assessment that addresses the treatment of sodium-bonded spent nuclear fuel.

Issues	*No. of Comments*	*DOE Responses*
This EIS is premature. The Draft SBSNF EIS should not be issued for public comment before publication of the National Academy of Science's Independent Assessment Final Report on the demonstration project, which probably won't be issued until October or November 1999. The National Academy of Sciences Final Report is answering the question, "Will it work," not, "Will it help?"	6	DOE believes that the results from the demonstration and the need to effectively utilize available resources justify the preparation of the EIS in parallel with the final demonstration reviews. The National Research Council has conducted ongoing reviews and issued status reports on the demonstration project. These reports are available for review and the final report will be considered when a record of decision is formulated.
DOE is premature in preparing this EIS because the demonstration project will not be completed until after the draft EIS is published.	11	The Electrometallurgical Treatment Research and Demonstration Project that began in June 1996 is scheduled to conclude in August 1999. At that time DOE will know if it has met the success criteria established by the National Research Council for the electrometallurgical treatment demonstration. DOE has obtained encouraging data from the demonstration to date, and is confident that the technology holds promise for the management of its sodium-bonded spent nuclear fuel inventory. Publication of the final report on the electrometallurgical treatment demonstration by the National Research Council may require a few months past the end of the demonstration project. DOE plans to make its decision in January 2000, based on the U.S. Nuclear Regulatory Commission final report and other factors such as cost, environmental consequences, and nonproliferation impacts.
DOE's willingness to proceed at this pace without even the completion of their demonstration project indicates the decision on pyroprocessing was made years ago.	2	DOE has made no decision on how the sodium-bonded spent nuclear fuel should be treated. The EIS addresses reasonable alternatives for treatment of this fuel.
More research and development should be completed before the Record of Decision on the alternatives.	1	DOE believes that enough is known about the alternatives to assess their environmental consequences in the EIS. DOE plans to make its decision on how to manage its sodium-bonded spent nuclear fuel in January 2000, based on such factors as technical feasibility, cost, environmental consequences, and nonproliferation impacts.
The EIS is premature in that there has not been enough time allowed to include the cost analysis.	1	A report comparing the costs of the alternatives will be made available to the public during the public comment period for the draft EIS.
We question the issuance of the Notice of Intent at this time and believe that it should be withdrawn pending compilation of all the technical documentation necessary to inform the scoping process.	1	DOE believes that adequate presentations, displays, and written materials on the proposed action and alternatives were provided to the public during the scoping process.
Although there is a regulatory driver for removal of this fuel from Idaho, that is not until 2035, and budget maintenance does not justify going ahead with this process until concerns about its technical feasibility, cost-effectiveness, and potential for proliferation have been adequately addressed. I recommend that DOE provide compelling evidence that it is prudent to proceed with preparing an EIS at this time.	2	DOE believes that enough is known about the alternatives to assess their environmental consequences in the EIS. DOE plans to make its decision on how to manage its sodium-bonded spent nuclear fuel in January 2000 based on factors such as technical feasibility, cost, environmental consequences, and nonproliferation impacts.

Issues	No. of Comments	DOE Responses
Miscellaneous		
This activity could be viewed as corporate welfare which, whether true or not, always is a concern.	2	DOE has identified the purpose and need for the proposed action, which is found in Section 1.2 of the draft EIS. Action is necessary for the responsible management of DOE's inventory of sodium-bonded spent nuclear fuel.
The intent of the agreement between the Governor of Idaho and DOE involves removing large amounts of radioactive materials, not just spent nuclear fuel.	1	The approximate 60 tons of sodium-bonded spent nuclear fuel currently stored in Idaho contains radioactive materials that cannot be reused, recycled, or disposed of in their current condition. Part of the intent of DOE's proposal is to prepare these materials for disposal or possible reuse for commercial purposes.
If a source is referenced in the EIS, it should be summarized in the EIS (e.g., EAR in the Depleted Uranium Hexafluoride Programmatic EIS).	1	Some reference documents are very large and difficult to summarize. Where practical, DOE has provided a brief summary of reference documents in the EIS.
DOE is not going to consider public comments; instead it is engaging in a show process that meets the bare minimum legal requirements.	1	DOE is considering and will continue to consider public comments in its sodium-bonded spent nuclear fuel management decision process. For example, DOE will provide a comparative Cost Study and a Nonproliferation Impacts Assessment to the public in response to comments received during the scoping process. Further, DOE has reformulated its proposed action in response to public comments.
It seems a bit of a waste of the public's time to continue to have these EISs in which we comment saying, "Slow down, we want more information," and DOE says, "Sure," and proceeds right along with its decision in the first place.	1	DOE is committed to providing the public the opportunity to review and comment on the proposed action to manage its inventory of sodium-bonded spent nuclear fuel.
This is not an EIS asking, "We've got a bunch of sodium-contaminated fuel. What should we do with it? We have the following five alternatives." We don't have an action that says, "We need to treat this fuel. We have EISs on it. We want to do pyroprocessing." It is lip service to the other alternatives that are available to deal with this spent nuclear fuel.	1	In response to public comments, DOE has revised the proposed action of the EIS from electrometallurgical treatment of sodium-bonded spent nuclear fuel in the Fuel Conditioning Facility at ANL-W to the treatment and management of sodium-bonded spent nuclear fuel.
We are gravely concerned with the project. We oppose it. We have opposed it all along.	1	DOE acknowledges the commentor's opposition to the proposed action.
That DOE is not waiting for the National Academy of Sciences' Final Report raises a question that Pit Nine also raises. DOE gets a lot of research and development money every year; do the data you collect mean anything?	1	The Electrometallurgical Treatment Research and Demonstration Project that began in June 1996 is scheduled to conclude in August 1999. At that time DOE will know if it has met the success criteria established by the National Research Council for the electrometallurgical treatment demonstration. DOE has obtained encouraging data from the demonstration to date, and is confident that the technology holds promise for the management of its sodium-bonded spent nuclear fuel inventory. Publication of the final report on the electrometallurgical treatment demonstration by the National Research Council may require a few months past the end of the demonstration project. DOE plans to make its decision in January 2000 based on the National Research Council's final report and other factors such as cost, environmental consequences, and nonproliferation impacts.

Issues	*No. of Comments*	*DOE Responses*
What is the endpoint for the National Research Council's waste characterization study? Is it a moving target or a dead horse?	1	The National Research Council is reviewing the waste qualification process and the acceptability of the waste forms.
I would like to see the products identified [cost analysis, nonproliferation analysis] in the briefing placed on a schedule that fits into the Secretary of Energy's decision on the Record of Decision. This schedule ought to be made available to the stakeholders.	1	DOE is preparing a Nonproliferation Impacts Assessment that addresses the treatment of sodium-bonded spent nuclear fuel. This assessment will be made available to the public during the draft EIS public comment period. DOE is also preparing a comparative Cost Study which will be made available to the public during the draft EIS public comment period.
In the past, DOE has had to redo work because of an inadequate initial assessment of a problem. The commentor hopes DOE will avoid such costly problems by proceeding only if it is clear that treatment is necessary. The commentor will be pleased to see DOE proceed with treating the spent nuclear fuel once adequate environmental documentation has been completed and once it has been established that treatment will be necessary before disposal.	1	This NEPA process will aid DOE in making an informed decision on how to proceed with the management of its sodium-bonded spent nuclear fuel. The alternatives analyzed in this EIS include no action and direct disposal with no treatment. DOE will make its decision in January 2000 based on the analytical results of this EIS combined with public comments on the draft EIS and the outcome of the demonstration project, as well as cost, schedule, and nonproliferation considerations.
Would it not be more realistic to base risk analysis on a Hormissis theory rather than the Linear Threshold theory?	1	The EIS acknowledges that there are other views on the effects of radiation at low dose rates. However, the linear dose response is the most accepted as well as the most conservative of current models, and is therefore appropriate for this analysis.
Press for the quickest, most scientifically proven solution to the preparation of this spent nuclear fuel for a repository.	1	DOE will make its decision in January 2000 based on the analytical results of this EIS combined with public comments on the draft EIS and the outcome of the demonstration project, as well as cost, schedule, and nonproliferation considerations.
Has integration/consolidation with other treatment/conditioning being performed at other DOE sites (Hanford, Savannah River) been considered?	1	DOE has considered the use of other DOE facilities as options for the management of sodium-bonded spent nuclear fuel. These issues were a major consideration of the DOE Programmatic Spent Nuclear Fuel EIS (April 1995). Alternatives 3 and 5 of the SBSNF EIS involve the use of two different facilities at SRS in South Carolina.
What happens in the No Action [Alternative] after 2035?	1	Under the No Action Alternative, the EIS evaluates the viability of direct disposal of sodium-bonded spent nuclear fuel in a geologic repository with no treatment, as well as storing the spent nuclear fuel and pursuing the research and development of a new or immature technology
Can the sodium be leached from the uranium?	1	The bond sodium could be melted and drained from the blanket fuel. The melt and drain process would not be effective on the sodium-bonded driver fuel because some of the bond sodium is inside or is encapsulated within the uranium material, and the uranium has become mechanically attached to the stainless-steel cladding.
Put the uranium into commercial fuel.	1	Although DOE has not made a decision regarding the disposition of low-enriched uranium, there is a possibility that the low-enriched uranium could be sold to the commercial reactor fuel industry as a feedstock material.

Issues	*No. of Comments*	*DOE Responses*
Few details about the [electrometallurgical treatment] process were provided [in the presentation].	1	The intent of the public scoping meeting presentation was to give the public a general overview of the NEPA process, electrometallurgical treatment, and other alternatives. The public meeting presentations during the draft EIS comment period will contain more detail about the electrometallurgical treatment process.
We believe that important questions about cost and waste characterization have been left out of most reviews of this program and urge the Energy Information Agency take an honest, comprehensive look at these issues.	1	As requested by members of the public during the scoping process, DOE is preparing a comparative Cost Report which will be made available to the public during the draft EIS comment period. DOE will make its decision in January 2000 based on the outcome of the demonstration project and other factors such as cost, environmental consequences, and nonproliferation impacts.
This program was featured on *NBC Nightly News* as a "Fleecing of America." According to DOE, this program is being created to cover the "redirection of valuable intellectual and physical resources at ANL.....as a result of the shutdown of the nuclear breeder reactor program known as the Advanced Liquid Metal Reactor). We are outraged that a key piece of a program that was supposedly terminated by Congress—the Advanced Liquid Metal Reactor—continues to squander taxpayer dollars on questionable "termination costs" and a wrong-minded "redirection" program known as pyroprocessing or electrometallurgical treatment at ANL. ...We are extremely concerned that this new "Nuclear Technology Research and Development" program represents nothing more than a continuation of the fuel reprocessing activities supported by the Advanced Liquid Metal Reactor program	1	The electrometallurgical treatment technology under consideration in the EIS for treating sodium-bonded spent nuclear fuel is a technology that was originally developed as part of DOE's Advanced Liquid Metal Reactor Program, which was discontinued in 1994. This technology was developed at significant expense to the taxpayer. DOE would be remiss in its responsibilities not to evaluate the potential application of this technology to the Department's sodium-bonded spent nuclear fuel. DOE believes that its proposal to apply electrometallurgical technology to the management of its sodium-bonded spent nuclear fuel inventory has the potential to solve a significant problem for the Nation.
DOE's record with other reprocessing technologies has been abysmal.	1	DOE has successfully used reprocessing technologies in the past to provide nuclear materials for research and defense purposes. The use of PUREX processing for the declad and cleaned blanket fuel [Alternative 3] is a viable option..
The [Snake River] Alliance encourages DOE to include ANL-W as part of INEEL in environmental analyses.	1	DOE has included the ANL-W facility as part of the INEEL in analyzing the environmental consequences of the alternatives in this EIS, as well as in the *DOE Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement.*
The commentor would prefer to see the spent nuclear fuel treated only once if possible.	1	DOE also would prefer to treat its sodium-bonded spent nuclear fuel only once, if at all, before its final disposition.
To support informed public review of the draft EIS, the schedule for this EIS should allow for adequate public review of related documents before the close of the public comment period.	1	The schedule for this EIS allows 45 days for public comment, in accordance with NEPA requirements. Related reports such as those on costs and nonproliferation issues will be available to the public within the same time frame as this draft EIS.

A.2 THE PUBLIC COMMENT PROCESS

A.2.1 Overview

In July 1999, DOE published the *Draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel.* NEPA regulations mandate a minimum 45-day public comment period after publication of a draft EIS to provide an opportunity for the public and other stakeholders to comment on the EIS analysis and results. The 45-day public comment period on the Sodium-Bonded Spent Nuclear Fuel (SBSNF) Draft EIS began on July 31, 1999, and was scheduled to end on September 13, 1999. In response to commentor requests, the comment period was extended an additional 15 days through September 28, 1999. During this 60-day comment period, public hearings were held in North Augusta, South Carolina; Boise and Idaho Falls, Idaho; and Arlington, Virginia (see **Figure A–2**). In addition, the public was encouraged to submit comments via the U.S. mail service, e-mail, a toll-free 800-number phone line, and a toll-free fax line. Section A.2.4 summarizes the major issues raised by comments received through the public comment process and DOE's position with respect to these comments.

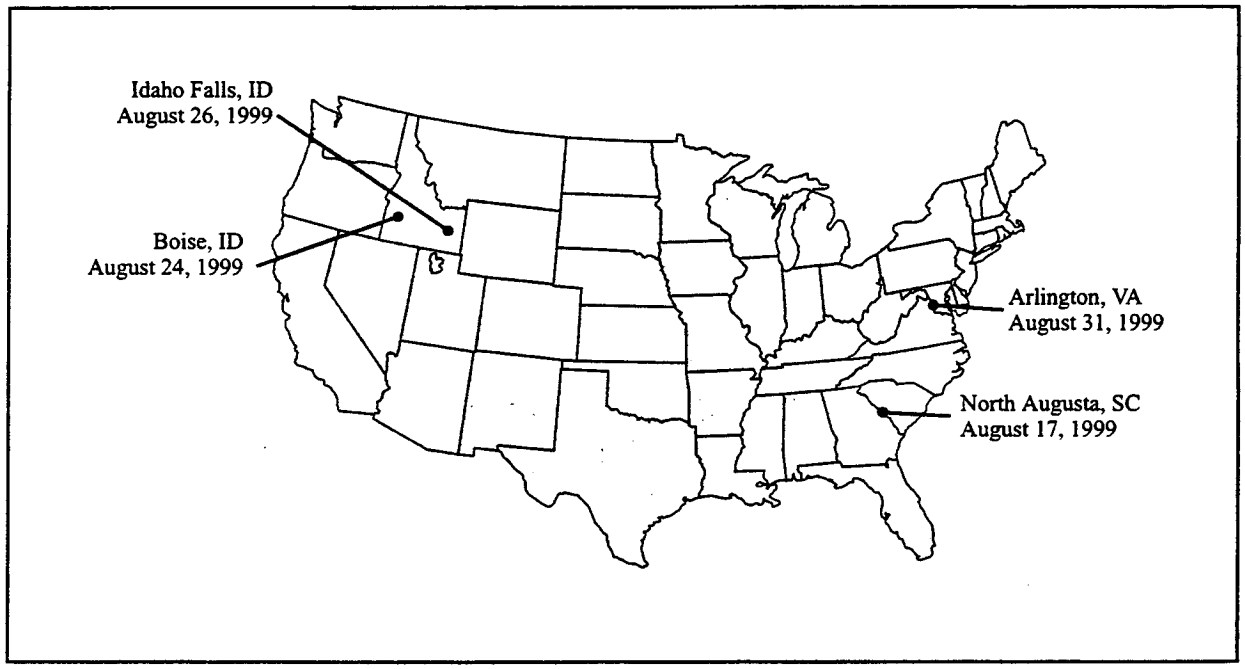

Figure A–2 Public Hearing Locations and Dates, 1999

The number of persons estimated in attendance at each hearing or meeting, together with the number of comments submitted and recorded, are presented in **Table A–4**. These attendance estimates are based on the number of registration forms completed and returned at each hearing or meeting, as well as a rough "head count" of the audience, and may not include all those present.

The public hearing comments were combined with comments received by other means (mail, e-mail, 800-number, fax) during the comment period. Written comments were date-stamped and assigned a sequential document number. **Table A–5** lists the number of comments received by method of submission.

Table A–4 Public Hearing/Meeting Locations, Attendance, and Comments Received

Location	*Date*	*Estimated Attendance*	*Comments*
North Augusta, South Carolina	August 17, 1999	20	18
Boise, Idaho	August 24, 1999	3	19
Idaho Falls, Idaho	August 26, 1999	45	21
Arlington, Virginia	August 31, 1999	20	25

Table A–5 Method of Comment Submission

Method	*Number of Comments*	*Number of Submittals*
Faxes	49	6
U.S. mail/hearing submittals	264	27
1-800 number	16	11
E-mail	82	12
Hearings (Number of Comment/Submittals)	83	16
Total Submittals	494	72

A.2.2 Public Hearing Format

The public hearings were organized to encourage public comments on the Sodium-Bonded Spent Nuclear Fuel (SBSNF) Draft EIS and to allow two-way interaction between public attendees and DOE representatives. A neutral facilitator was present at each hearing to direct and clarify discussions and comments. A court reporter also was present at each hearing to record the proceedings and provide a transcript of the public comments and the dialogue between the public and the DOE and contractor representatives on hand. These transcripts are available in DOE public reading rooms near each of the proposed sites and in Washington, D.C.

The format used for each hearing included a presentation, question and answer session, and a public comment period. The hearing opened with a welcome from the facilitator, followed by a presentation on the proposed action by a DOE representative. The facilitator next opened the question and answer session to give the audience a chance to ask questions about the material presented. This was followed by the public comment session, during which attendees were given an opportunity to read a prepared statement of no more than five minutes. Modifications to the format were made at each of the public hearings to fulfill the special requests of attendees. Following the public hearings, the comments were identified from the transcripts of each hearing and the comment documents submitted by the attendees.

A.2.3 Comment Disposition

Comments received at the public hearings and via fax, U.S. mail, e-mail, or the toll-free 800-number phone line were divided into ten issue categories to facilitate responses and provide an overview of the type of comments that DOE received. The categories appear in Table A–8 later on in this appendix.

All the comments received during the SBSNF Draft EIS comment period appear in either Section A.2.5 or A.2.6 of this appendix. Section A.2.5 contains a set of tables corresponding to each of the public hearings. Section A.2.6 includes scanned images of the comments received via U.S. mail, e-mail, toll-free phone line, toll-free fax line, or personal submission at the public hearings. DOE's response to each comment is presented

on the opposite side of the page. Transcriptions of the oral comments submitted at each of the public hearings are presented in the appropriate tables, along with DOE's responses to each comment.

Table A–6 is an index of all of the commentors who made statements or submitted comments at the public hearings or during the public comment period, including members of the public, representatives of organizations or agencies, and public officials. Commentors are listed alphabetically by their last name, along with the page on which their comments appear in Sections A.2.5 or A.2.6. **Table A–7** identifies separately Federal, State, and local officials and agencies, companies, organizations and special interest groups that submitted comments. **Table A–8** correlates comment categories with comment identification numbers; thus, permitting the reader to readily locate similarly categorized comments.

Table A–6 Commentors Index

Commentor	*Commentor Number*	*Comment/Response Page Numbers*
David E. Adelman, Natural Resources Defense Council, Washington, DC	36	A-140
Richard Albrecht, Wilson, WY	2	A-76
Anonymous	18	A-112
Anonymous	15	A-92
Anonymous	19	A-113
Robert Bobo, The Shoshone-Bannock Tribes, Fort Hall, ID	55	A-209
Charles Bailey	6	A-80
Julie Bowles, Boise, ID	40	A-148
Jean Boyles	7	A-81
Beatrice Brailsford, Snake River Alliance, Pocatello, ID	706	A-57
Ted L. Carpenter, The Shoshone-Bannock Tribes, Fort Hall, ID	47, 703	A-172, A-54
Ernest S. Chaput, Economic Development Partnership of Aiken and Edgefield Counties of South Carolina, Aiken, SC	13, 504	A-88, A-42
Pat Clark, Snake River Alliance, Boise, ID	5	A-79
John Commander, Coalition 21, Idaho Falls, ID	27, 56, 701	A-125, A-213, A-52
Peter J. Dirkmaat, DOE-ID, Shelley, ID	3	A-77
dpdufur@micron.net	21	A-116
Beth Duke, Sun Valley, ID	20	A-114
Maureen Eldredge, Alliance for Nuclear Accountability, Washington, DC	800	A-59
Nancy Fenn	25	A-122
Dan Freeman	39	A-147
Rick Gheddis	502	A-38
Ellen Glaccum, Ketchum, ID	1	A-73
Kathryn Graves, Hailey, ID	44	A-159
Jeep Hardinge, Ketchum, ID	12	A-87
David Hensel, Driggs, ID	31	A-131
Steve Herring, Idaho Section of ANS, Idaho Falls, ID	704	A-55
Steve Hopkins, Snake River Alliance, Boise, ID	17, 41, 600	A-111, A-149, A-44
Laird Irvin, Ketchum, ID	9	A-83
Lowell Jobe, Coalition 21, Idaho Falls, ID	8, 32, 56	A-82, A-133, A-213

Commentor	Commentor Number	Comment/Response Page Numbers
Lisa Johnson, Victor, ID	33	A-134
Dan Johnston, Richland, WA	34	A-137
Dick Kenney, Coalition 21, Idaho Falls, ID	702	A-53
David Kipping, Snake River Alliance, Boise, ID	30	A-128
Lisa Ledwidge, Institute for Energy & Environmental Research, Takoma Park, MD	46	A-162
Edwin Lyman, Nuclear Control Institute, Washington, DC	52, 802	A-196, A-68
Susan Mathees, Ketchum, ID	11	A-86
Barbara Mathison, Meridian, ID	54	A-207
Betina Mattesen, Bristol, VT	10	A-84
Patricia McCracken, Augusta, GA	16, 503	A-93, A-39
Don McWhorter, North Augusta, SC	14	A-90
Carol Murphy, Ketchum, ID	35, 37, 39	A-138, A-145, A-147
Susan Pengilly Neitzel, Idaho State Historical Society, Boise, ID	4	A-78
Suzy Nielond, Jackson, WY	38	A-146
Richard Parkin, U.S. Environmental Protection Agency, Seattle, WA	53	A-202
Debra Patla, Victor, ID	48	A-173
Lee Poe, Aiken, SC	500	A-35
Randy Ponic	501	A-37
Bennett Ramberg, Committee to Bridge the Gap, Los Angeles, CA	50	A-185
Charles Rice, INEEL Citizens Advisory Board, Idaho Falls, ID	51	A-191
Matt Smith	23	A-118
Margaret Stewart, Ketchum, ID	42	A-154
John Tanner, Coalition 21, Idaho Falls, ID	26, 705	A-124, A-56
Willie R. Taylor, U.S. Department of Interior, Washington, DC	43	A-157
Marlise Teasley, Twin Falls, ID	45	A-160
Kathleen E. Trever, State of Idaho INEEL Oversight Program, Boise, ID	49	A-177
Doug Turner, Bechtel Jacobs Company LLC, Oak Ridge, TN	22	A-117
Robert H. Wilcox, Martinez, GA	29	A-127
Terry & Theresa Williams, Hailey, ID	28	A-126
Monte Wilson, Potlatch, ID	24	A-120
Hisham Zerriffi, Institute for Energy & Environmental Research, Takoma Park, MD	46, 801	A-162, A-61

Table A–7 Index of Public Officials, Organizations, and Public Interest Groups

Commentor Information	*Document Number*	*Page Number*
Alliance of Nuclear Accountability, Maureen Eldredge, Washington, DC	800	A-59
Coalition 21, Idaho Falls, ID	8, 26, 27, 32, 701, 702, 705	A-82, A-124, A-125, A-133, A-52, A-53, A-56
Committee to Bridge the Gap, Benett Ramberg, Ph.D., Director of Research, Los Angeles, CA	50	A-185
Economic Development Partnership of Aiken and Edgefield Counties of South Carolina, Ernest Chaput, Aiken, SC	13, 504	A-88, A-42
Idaho State Historical Society, Susan Pengilly Neitzel, Deputy State Historic Preservation Officer and Compliance Coordinator, Boise, ID	4	A-78
INEEL Citizens Advisory Board, Charles Rice, Chair, Idaho Falls, ID	51	A-191
Institute for Energy & Environmental Research, Hisham Zerriffi, Project Scientist, and Lisa Ledwidge, Outreach Coordinator, Takoma Park, MD	46, 801	A-162, A-61
Natural Resources Defense Council, David E. Adelman, Project Attorney, Washington, DC	36	A-140
Nuclear Control Institute, Edwin Lyman, Scientific Director, Washington, DC	52, 802	A-196, A-68
Shoshone-Bannock Tribes, Robert Bobo, Project Director, and Ted Carpenter, DOE Project Environmentalist, Fort Hall, ID	47, 55, 703	A-172, A-210, A-54
Snake River Alliance, David Kipping, President, Board of Directors, and Steve Hopkins, Program Assistant, Boise, ID	17, 30, 41, 600, 706	A-111, A-128, A-149, A-46, A-57
State of Idaho INEEL Oversight Program, Kathleen Trever, Coordinator-Manager, Boise, ID	49	A-177
U.S. Department of the Interior, Office of the Secretary, Willie Taylor, Director, Office of Environmental Policy and Compliance, Washington, DC	43	A-157
U.S. Environmental Protection Agency, Region 10, Richard Parkin, Manager Geographic Implementation Unit, Seattle, WA	53	A-203

Table A–8 Comment Categories and Comment Identification Numbers

Comment Categories	*Comment Identification Numbers*
1.0 Purpose, Need for, and Timing of Proposed Action	1-4, 16-26, 16-62, 16-77, 17-2, 25-11, 27-3, 27-5, 31-8, 35-2, 41-2, 41-3, 41-13, 45-2, 46-3, 46-4, 46-7, 46-8, 46-11, 46-13, 47-3, 48-4, 52-3, 52-4, 53-1, 55-4, 55-8, 600-7, 600-8, 600-14, 702-4, 800-2, 800-3, 800-4, 800-6, 801-3, 801-4, 801-7, 801-8, 801-9, 801-11
2.0 Waste Disposition, Waste Acceptance Criteria	10-1, 10-6, 14-1, 14-2, 16-6, 16-8, 16-13, 16-14, 16-22, 16-23, 16-24, 16-27, 16-51, 16-52, 19-1, 20-6, 24-5, 25-2, 25-10, 26-4, 30-7, 31-6, 33-3, 33-10, 35-3, 36-10, 39-5, 41-8, 41-9, 41-11, 42-5, 46-6, 48-6, 49-4, 49-8, 49-24, 49-25, 49-26, 49-28, 49-29, 49-35, 49-36, 51-9, 52-7, 54-3, 55-7, 56-7, 500-6, 600-10, 705-4, 801-2, 801-6, 801-10, 802-3, 802-8
3.0 NEPA and Extension of Public Comment Period	1-1, 1-2, 1-3, 5-1, 7-1, 8-1, 8-3, 9-1, 10-3, 11-1, 12-1, 16-2, 16-3, 16-17, 16-34, 16-35, 16-39, 16-41, 16-44, 16-45, 16-65, 16-78, 19-2, 20-1, 20-2, 21-1, 23-1, 24-1, 25-5, 28-1, 29-1, 29-2, 30-1, 30-2, 30-8, 31-1, 32-1, 33-6, 35-5, 35-7, 36-1, 36-3, 36-4, 36-7, 36-14, 37-2, 38-1, 39-1, 39-2, 41-1, 41-5, 41-7, 42-1, 42-2, 42-6, 42-8, 42-9, 43-3, 44-4, 45-3, 48-10, 49-1, 49-10, 49-12, 49-17, 49-21, 49-31, 49-32, 49-39, 51-1, 51-4, 51-8, 51-10, 52-1, 53-2, 54-5, 55-1, 56-1, 56-12, 503-4, 600-1, 600-2, 600-3, 600-4, 600-6, 600-12, 706-2, 706-3, 800-1, 802-1

Comment Categories		Comment Identification Numbers
4.0	Relationship to other DOE Programs	1-7, 16-19, 16-25, 16-28, 16-29, 16-31, 16-32, 16-40, 16-50, 16-64, 23-2, 23-6, 24-2, 25-6, 29-4, 30-3, 31-2, 33-7, 35-6, 41-4, 41-12, 42-7, 44-2, 45-1, 46-5, 46-10, 49-5, 49-6, 49-27, 51-7, 54-6, 54-10, 503-1, 702-3, 801-5, 802-4
5.0	Out of Scope - Cost	10-4, 13-5, 15-1, 16-7, 16-9, 16-11, 16-12, 16-20, 16-30, 16-42, 16-43, 16-46, 16-48, 16-49, 16-55, 16-57, 16-58, 16-59, 20-5, 23-4, 25-4, 25-8, 29-3, 29-6, 30-5, 31-4, 31-9, 32-2, 33-4, 36-8, 36-9, 36-12, 37-4, 39-4, 40-2, 42-4, 48-8, 51-5, 54-4, 54-8, 56-3, 56-4, 56-5, 56-6, 504-4, 600-15, 700-1, 802-2
6.0	Out of Scope - Nuclear Nonproliferation Policy	10-2, 17-1, 20-4, 23-5, 24-4, 25-3, 25-9, 26-3, 27-4, 30-6, 31-5, 31-7, 33-5, 33-9, 35-4, 41-6, 41-15, 44-1, 46-1, 46-16, 46-17, 46-18, 46-19, 46-20, 46-21, 46-22, 48-3, 50-1, 51-6, 52-8, 52-9, 52-10, 52-11, 52-12, 52-13, 52-14, 54-2, 54-9, 56-11, 501-1, 600-5, 600-13, 600-17, 700-3, 701-3, 801-12
7.0	Technologies (Technical Issues)	13-4, 14-3, 14-4, 14-5, 16-10, 16-15, 16-16, 16-18, 16-36, 26-1, 34-2, 36-5, 46-2, 46-9, 46-12, 47-1, 49-7, 49-9, 49-37, 55-5, 55-6, 500-5, 504-3, 700-2, 703-1, 705-1, 705-2, 802-5
8.0	Alternatives (NEPA-Related Issues)	2-1, 3-2, 6-1, 13-1, 13-2, 13-3, 16-21, 16-33, 18-1, 20-3, 22-1, 24-6, 24-7, 26-2, 27-1, 27-2, 28-2, 29-5, 33-2, 35-1, 36-6, 36-11, 36-13, 37-1, 39-3, 40-1, 41-10, 41-14, 44-3, 46-23, 48-1, 49-2, 51-2, 51-3, 52-5, 54-1, 55-2, 55-9, 56-2, 56-8, 56-9, 56-10, 500-2, 500-3, 500-4, 504-5, 502-1, 504-1, 504-2, 600-16, 600-18, 600-19, 701-1, 701-2, 701-4, 702-2, 704-1, 705-5, 706-1, 801-1, 802-6, 802-7
9.0	Affected Environment/Environmental Consequences	1-5, 1-6, 1-8, 3-1, 4-1, 16-1, 16-4, 16-5, 16-37, 16-38, 16-47, 16-53, 16-54, 16-56, 16-60, 16-61, 16-63, 16-66, 16-67, 16-68, 16-69, 16-70, 16-71, 16-72, 16-73, 16-74, 16-75, 16-76, 23-7, 33-1, 34-1, 40-3, 43-1, 43-2, 43-4, 46-14, 46-15, 47-2, 48-7, 48-9, 49-3, 49-11, 49-13, 49-14, 49-15, 49-16, 49-18, 49-19, 49-20, 49-22, 49-23, 49-30, 49-33, 49-38, 49-40, 49-41, 49-42, 49-43, 52-6, 53-3, 53-4, 53-5, 53-6, 53-7, 53-8, 53-9, 55-3, 500-1, 503-3, 503-5, 702-1, 703-2, 800-5
10.0	Out of Scope - Other	10-5, 21-2, 23-3, 24-3, 25-7, 30-4, 31-3, 33-8, 36-2, 42-3, 48-11, 54-7, 503-2, 600-9, 600-11

A.2.4 Issues Raised During the Public Comment Period

Four hundred and ninety-four comments were received during the public comment period. Most of the comments focused on the following: (1) the purpose, need for, and timing of the proposed action; (2) the introduction of new waste forms produced by the proposed action, their acceptability in a geologic repository, and the disposition of uranium and plutonium by-products; (3) the public availability of information considered relevant to reviewing the draft EIS, the extension of the comment period, and the relationship of the EIS to other DOE programs; (4) the cost of the various alternatives; (5) the impacts of the proposed action on U.S. nuclear nonproliferation policy; (6) technical and/or NEPA-related questions regarding technologies and alternatives; and (7) questions related to the affected environment and the environmental consequences. DOE's responses to these issues are summarized below. The comments also dealt with a number of other subjects, including technologies considered and dismissed from further evaluation, long-term (beyond institutional control) performance of the sodium-bonded spent nuclear fuel during storage on site, and questions on the methodology and assumptions of the health and safety analysis. Many commentors expressed their opposition or support for DOE's action in general or for specific alternatives under the proposed action or the No Action Alternative. Section A.2 of Appendix A provides DOE's responses to all comments on a comment-by-comment basis.

Purpose, Need for, and Timing of the Proposed Action

Many comments expressed the opinion that DOE failed to demonstrate the purpose and need for the proposed action or to provide a rationale for its timing. Some of the reasons given included the lack of a compelling argument that there is a safety risk associated with current storage; the lack of a regulatory framework and final waste acceptance criteria; the lack of an approved site for a geologic repository; insufficient information on the results of the Electrometallurgical Treatment Research and Demonstration Project; and the lack of analysis showing that direct disposal of the sodium-bonded spent nuclear fuel without sodium removal would be detrimental to the performance of the geologic repository.

DOE's position as presented in the EIS is that the need to examine options for the management and treatment of sodium-bonded spent nuclear fuel is based on the existing regulatory environment concerning long-term disposal of spent nuclear fuel and high-level radioactive waste. It is assumed that DOE's sodium-bonded spent nuclear fuel, as well as other DOE-owned spent nuclear fuel, eventually will be disposed of in a geologic repository. One of the key requirements, as specified in the current April 1999 version of the DOE'S *Waste Acceptance Systems Requirements Document* and in the U.S. Nuclear Regulatory Commission requirements for acceptance of spent nuclear fuel or high-level waste in a geologic repository, is that it cannot contain or generate materials that are explosive, pyrophoric, or chemically reactive in a form or amount that could compromise the repository's ability to perform its waste isolation function or to satisfy its performance objective (10 CFR 60.135(b)(1)). The sodium-bonded spent nuclear fuel, if left in its existing state, would contain pyrophoric and chemically reactive metallic sodium and, therefore, would not likely meet DOE or U.S. Nuclear Regulatory Commission repository acceptance criteria.

The timing for the proposed action is a programmatic issue rather than a safety issue. The EIS does not conclude that current storage of sodium-bonded spent nuclear fuel presents a threat to the health and safety of workers or the public. The programmatic risk associated with implementing the proposed action or not treating the sodium-bonded spent nuclear fuel is the uncertainty surrounding the acceptability of this fuel for placement in a geologic repository. The process of establishing a repository is dependent on not only the site but also the materials to be disposed of. As part of this process, a total system performance assessment that describes the probable behavior of a repository is performed. This total system assessment includes the performance of the specific waste forms and inventories proposed for disposal. As part of the process of establishing a repository, data for the waste forms are needed prior to making a final selection of the repository, not after. In fact, if specific waste forms are not represented in crucial documents like this EIS, additional documentation will be needed to allow for the possibility of disposing of those materials in the repository. The performance of sodium-bonded spent nuclear fuel in a geologic repository depends on many factors (e.g., long-term fuel integrity, repository environment fuel/waste package survivability, etc.), and the presence of metallic sodium would complicate the modeling even further. Stabilization of the spent nuclear fuel and/or removal of the metallic sodium would provide greater protection for human health and the environment.

The Electrometallurgical Treatment Research and Demonstration Project began in June 1996 and, although the review of the test results has not been finalized in a single report, a number of status reports were issued by DOE and reviewed by the National Academy of Sciences' National Research Council Committee. They are referenced in the EIS. The success criteria established at the outset of the project have been fulfilled. The environmental impact analysis associated with the electrometallurgical treatment process alternatives was based on actual data from the demonstration project. The final EIS includes a new section on the status and results of the project. Having completed the demonstration project and in planning the closure of its PUREX processing capabilities, DOE now needs to decide whether these processes are suitable for treating the remaining sodium-bonded spent nuclear fuel, or whether there is sufficient reason to delay a decision and wait for the development of other treatment technologies. Delaying the EIS could result in a loss of capability and

of experienced, knowledgeable technical staff, should DOE decide at a later date to use the electrometallurgical process to treat the sodium-bonded spent nuclear fuel.

New Waste Forms and Disposition of Uranium and Plutonium By-Products

Some of the comments questioned the generation of new waste forms from treating the sodium-bonded spent nuclear fuel and the possible acceptance of these forms in a geologic repository. Also, a number of commentors remarked on the generation of uranium and plutonium as by-products of the treatment process. Related issues were the disposition of uranium metal, a by-product of the electrometallurgical process, and the compliance of both the PUREX and the electrometallurgical process with U.S. nuclear nonproliferation policy in terms of the separation of these elements.

All of the alternatives evaluated in this EIS would produce some form of high-level radioactive waste. Electrometallurgical treatment would produce two new waste forms (metallic, ceramic) and the melt and dilute process would produce a new metallic form (i.e., melt and dilute product or conditioned spent nuclear fuel). These forms would be more stable than the untreated sodium-bonded spent nuclear fuel. The ceramic and metallic waste forms generated during the electrometallurgical treatment process represent chemically stable materials compared to untreated sodium-bonded spent nuclear fuel. The production of a chemically stable waste form to replace a chemically reactive waste form (i.e., sodium-bonded spent nuclear fuel) represents an improvement in the safe, long-term storage of this spent nuclear fuel. DOE expects the new waste forms to be suitable for disposal in a repository and to meet the requirements of the final waste acceptance criteria. The high-level radioactive waste form resulting from the PUREX process is borosilicate glass, which has been extensively tested and analyzed under conditions relevant to a geologic repository.

With respect to uranium and plutonium disposition, the EIS states that only uranium that would be separated under the electrometallurgical process would be blended down and stored on site if it originates from driver spent nuclear fuel, or would be stored on site as depleted uranium if it originates from blanket spent nuclear fuel. The final disposition of the stored uranium has not been decided and is not discussed in the EIS. The disposition of the uranium will be subject to a separate NEPA review. The nuclear nonproliferation policy aspects of this separation is subject to the nuclear nonproliferation policy assessment of the alternatives. The approximately 260 kilograms (575 pounds) of plutonium that would be separated under the PUREX process would be disposed of in accordance with the Record of Decision (65 FR 1608) for the *Surplus Plutonium Disposition Environmental Impact Statement* (DOE/EIS-0283) issued in November 1999. This separation is the subject of the nuclear nonproliferation assessment, which is independent of this EIS.

Public Availability of Information and Related Documentation

Many commentors asked for a 60-day extension of the 45-day public comment period on the draft EIS. Commentors said they wanted additional time to obtain and review relevant documents such as the Yucca Mountain Draft EIS and the National Academy of Sciences' National Research Council's final report on the Electrometallurgical Treatment Research and Demonstration Project, as well as the Cost Study and Nonproliferation Impacts Assessment. The comments frequently stated that DOE needs to make all of this information publicly available before the end of the EIS comment period and the issuance of the final EIS and the Record of Decision.

In an effort to ensure that all interested parties had time to comment on the draft EIS, the due date for transmittal of comments was extended from September 13 to September 28, 1999 (64 FR 49169). With respect to the need for more information, DOE made that information available to the public. Background materials were placed in public reading rooms and were made available to the public through a series of

hearings held August 17, 1999, in North Augusta, South Carolina; August 24, 1999, in Boise, Idaho; August 26, 1999, in Idaho Falls, Idaho; and August 31, 1999, in Arlington, Virginia. Materials placed in the reading rooms included the electrometallurgical demonstration environmental assessment, the Finding of No Significant Impact for the environmental assessment, National Research Council reports, the 1995 Settlement Agreement and Consent Order with the State of Idaho, the scoping meeting transcripts and comments, and the draft EIS hearing presentations and fact sheets. In addition, completion of the Cost Study and Nonproliferation Impacts Assessment was expedited so that they would be available to the public at the beginning of the comment period. These reports were mailed to interested parties on August 12, 1999, and were made available to attendees at all of the public hearings on the draft EIS. Although these reports are not critical to the evaluation of the analysis presented in the draft EIS, they will provide input to the Record of Decision. While the final National Research Council report on the demonstration project was published in April 2000, interim status reports were produced throughout the project. Data generated during the demonstration project were used in preparing the EIS.

Cost Issues

A number of commentors raised cost issues and provided comments directly related to the Cost Study, which was not part of the EIS.

Comments concerning the costs of the proposed action were considered beyond the scope of the EIS. The EIS was prepared in accordance with NEPA, as well as the Council on Environmental Quality's regulations on implementing NEPA (40 CFR 1500 through 1508) and DOE's NEPA regulations (10 CFR 1021). None of these regulations require the inclusion of a cost analysis in an EIS. The basic objective of the SBSNF EIS is to provide the public and DOE decision-makers with a description of the reasonable alternatives for treating and managing sodium-bonded spent nuclear fuel and information about their potential impacts on public health and safety and the environment. While cost could be an important factor in the ultimate Record of Decision, the purpose of this and other EISs is to address the environmental consequences of the proposed action and the No Action Alternative. DOE distributed cost information through the independent Cost Study released in August 1999, and this information is available to the public on request and in the DOE's public reading rooms. Responses to specific comments related to cost issues are included in Sections A.2.5 and A.2.6 of this appendix.

Nuclear Nonproliferation Policy Issues

The nuclear nonproliferation implications of the proposed action were the subject of a number of comments. Some commentors expressed strong opinions about how the use of specific technologies such as electrometallurgical treatment might impact U.S. nonproliferation policy.

Nonproliferation is another issue that was considered beyond the scope of the EIS. A separate Nonproliferation Impacts Assessment was prepared by DOE's Office of Arms Control and Nonproliferation. After assessing the potential nonproliferation impacts that may result from each of the alternatives and technologies analyzed in the SBSNF Draft EIS, the Office of Arms Control and Nonproliferation found that all the alternatives, except that involving PUREX processing at Savannah River, are fully consistent with U.S. policy concerning reprocessing and nuclear nonproliferation. Electrometallurgical treatment, for example, would not increase national inventories of weapons-usable fissile material because, although highly enriched uranium is an interim product of the process, it would be blended down to low-enriched uranium during treatment. Within the current equipment configuration and design, it is not possible to produce weapons-usable plutonium merely by adjusting the operating parameters. To do this, traditional aqueous processing would be required after electrometallurgical treatment. However, traditional aqueous processing could be used to produce weapons-usable plutonium directly from the spent nuclear fuel, without

electrometallurgical treatment, so electrometallurgical treatment itself does not present a special proliferation concern. Responses to specific comments related to nonproliferation are included in Sections A.2.5 and A.2.6 of this appendix.

Technologies, Alternatives

Various comments dealt with technical questions and issues regarding the treatment technologies addressed in the EIS or NEPA-related issues regarding the selected alternatives.

The variety of the issues precludes a summary response. Responses to these questions on a comment-by-comment basis are included in Sections A.2.5 and A.2.6 of this appendix. A number of the responses indicate that revisions to the EIS were made as a result of the comments.

Affected Environment and Consequences

A number of comments included questions concerning the description of the affected environment in the SBSNF Draft EIS, and the results of the environmental impact analysis.

As in the case above, responses to these comments on a comment-by-comment basis are included in Sections A.2.5 and A.2.6 of this appendix.

A.2.5 Public Hearing Comments and DOE Responses

Comments presented in this section were submitted during oral presentations at the public hearings held on August 17, 1999, in North Augusta, South Carolina; August 24, 1999, in Boise, Idaho; August 26, 1999, in Idaho Falls, Idaho; and August 31, 1999, in Arlington, Virginia. DOE's responses to these comments are also presented.

Comments from the North Augusta, South Carolina, Public Hearing August 17, 1999		
No.	*Comment*	*DOE Response*
Lee Poe		
500-1	"In your charts you show the maximum potential radiological impacts...that the PUREX process has those rates that exceed background. It just seems unreasonable...knowing the canyons and their operations like they do. Would you explain how you got a dosage of one and a half times background?" *[The commentor is referring to DOE's presentation of the worker dose at SRS of 500 millirem per year compared to a background dose of 360 millirem per year.]*	The average SRS worker dose used to evaluate environmental impacts is routinely assumed to be 500 millirem per year. This dose value is conservative and has been published in numerous environmental impact statements on SRS. As indicated in Section E.4.3 of the EIS, this average worker dose estimate was also used in the SRS Spent Nuclear Fuel Management EIS for activities similar to those described in this SBSNF EIS.
500-2	"I notice that when you showed the pictures of the alternatives, all but one of the drivers are processed through the electrorefining process at INEEL ANL-West. That was a surprise to me, that there were no other alternatives other than the melt and dilute."	Technologies such as GMODS and the direct plasma arc-vitreous ceramic processes have the potential to be used to treat driver sodium-bonded spent nuclear fuel. However, as discussed in Section 2.6 of the EIS, these technologies are less mature than those evaluated in detail in the EIS.
500-3	"If we've got a technology that's marginal, is there something out there that will mature in the next 10 years that would allow that material to be processed? ...I think that's an issue you need to address more than what I saw. Now, maybe it's addressed in there, but what I saw was those alternatives were fairly written off."	As discussed in Section 2.5.1, the EIS evaluates two options under the No Action alternative: (1) direct disposal of the sodium-bonded spent nuclear fuel without sodium removal, and (2) continued storage until 2035 in its current location or until a technology, currently dismissed as less mature, is developed. From an environmental point of view, the development of a promising technology could require a considerably long time (20 to 30 years) and would still have to be viable to complete treatment of all or part of the sodium-bonded spent nuclear fuel before 2035.

Comments from the North Augusta, South Carolina, Public Hearing August 17, 1999		
No.	***Comment***	***DOE Response***
500-4	"The one thing that's different in the No Action is that you didn't analyze failure of the material...as spent fuel storage...way out into the future as the repository has done for that material. And if you don't bury it.... If it doesn't go to the mountain and stays at Idaho or wherever, you know, wherever DOE wants to put it, what's the consequence of No Action? And I would think that ought to be more clearly analyzed in the document."	Normal operation radiological effluent from potential fuel degradation during storage at INEEL up to 2035 is evaluated under the No Action Alternative in Section 4.2 of the EIS. As discussed in revised Section 2.5.1 of the EIS, a fundamental assumption made under the No Action Alternative is that sodium-bonded spent nuclear fuel would be disposed of in a repository along with the rest of the DOE-owned spent nuclear fuel within a finite period of time while under the institutional control of DOE. This EIS covers a time period up to 2035, at which time sodium-bonded spent nuclear fuel stored in Idaho would have to be transported out of the state and either stored or treated at another DOE site. For such an eventuality, additional NEPA documentation would be required. The unlikely scenario that treated sodium-bonded spent nuclear fuel would remain at its current site beyond 2035 because there is no geologic repository to accept it has been evaluated as part of the No Action Alternative in the Yucca Mountain Draft EIS, which was issued by DOE in July 1999. The Yucca Mountain EIS is discussed in Section 1.6.2.2.
500-5	"I think of melt and dilute as being a process that you need to isotopically dilute the uranium in the driver fuel. I wonder why you call it melt and dilute. It would seem like to me it's melt and—you know, it's not melt and dilute, then, so you ought to call it by a name that's appropriate. I understand that it's using the equivalent. You may be saying dilute it with aluminum but, you know, that's not clear to the — to the reader from the EIS as to what it is that makes it called melt and dilute."	The melt and dilute process described in the EIS is consistent with the general definition; i.e., it produces a larger volume and a lower concentration by adding material fillers (aluminum, stainless steel, or uranium metal).

Comments from the North Augusta, South Carolina, Public Hearing August 17, 1999		
No.	*Comment*	*DOE Response*
500-6	"I'm terribly disappointed to see that the progress of getting disposal criteria, Waste Acceptance Criteria, for the various fuel other than the commercial power reactor fuel has been almost nonexistent. It certainly appears from reading the Yucca Mountain EIS that...the high-level waste is...way ahead of the government spent nuclear fuel, our stepchildren, and they don't have...anybody there driving it....I would encourage the DOE folks to get out there and to get the DOE spent nuclear fuel, whatever it takes, to get the WAC requirements for those. And if that means a different level of treatment than we're all thinking about or if it means something else, then we ought to be working in that direction. Let's don't stabilize it twice. Let's don't do it now and then turn around 10 years from now and, when it comes time, they open the mountain and all of a sudden they say, 'Ah, you don't have any requirements for that.' So to the DOE folks, let me encourage you to do whatever you can to force RW into working with you to get specifications for waste disposal."	The borosilicate glass waste form for the PUREX alternative has been extensively tested and analyzed under conditions relevant to a geologic repository. One objective of the Electrometallurgical Treatment Research and Demonstration Project was to characterize the electrometallurgical treatment waste forms to facilitate their acceptability in a geologic repository. To ensure the treatment option that might be selected by DOE would produce a product that is likely to meet the acceptance criteria, DOE is working with the National Research Council to obtain comments on the research and development activities DOE will perform to establish treatment technology specifications. The EIS discusses the status of the waste acceptance criteria in Section 2.7 and the environmental impacts of the No Action Alternative in Section 4.2. The timing of DOE's decision on the treatment and management of sodium-bonded spent nuclear fuel in relation to the availability of a geologic repository is discussed in Section 4.12.2.
Randy Ponic		
501-1	"I was looking at the nonproliferation study to support this and one of the comments was they found the canyon operations in this report to be somewhat inconsistent with nonproliferation policy. Yet, in a similar report that was done for the melt and dilute process, they did not find that inconsistency. They found that the canyon operations would be consistent with policy. And using this report actually biases the canyon operations as far as this alternative. So that needs to be addressed, why there's reliance here and not in the previous report that was done for dealing with clad fuels."	The assessment of nonproliferation impacts is not part of the scope of the EIS. However, the "Nonproliferation Impact Assessment for the Management of the Savannah River Site Aluminum-Based Spent Nuclear Fuel" stated that use of conventional reprocessing (PUREX processing) to mitigate safety and health vulnerabilities is consistent with U.S. policy on plutonium reprocessing and the use of plutonium. Since safety and health vulnerabilities do not currently exist for the sodium-bonded spent nuclear fuel, use of conventional reprocessing (PUREX) in this case is somewhat inconsistent with U.S. nonproliferation policy. In this instance, the inconsistency would be due to the generation of potentially usable weapons-grade plutonium. The plutonium product from PUREX processing would be addressed by the Surplus Plutonium Disposition EIS.

Comments from the North Augusta, South Carolina, Public Hearing August 17, 1999		
No.	***Comment***	***DOE Response***
Rick Gheddis		
502-1	"It seems ... strange that the melt and dilute at SRS is not applied for the driver fuels. Its design is an HEU treatment process, yet you're applying it only on the blanket fuels, which are depleted uranium, and it's not particularly well suited for depleted uranium operations. Therefore, I'd like to make a comment that you consider an alternative of melt and dilute on the driver fuels at SRS. And by the way, I'd like to see that paired up with the PUREX processing of the blanket fuels, see that as an area of alternative...the blanket fuels match up very well with the PUREX processing."	The commentor's preference for the treatment of both driver and blanket sodium-bonded spent nuclear fuel at SRS is noted. As a result of the commentor's remarks, the possibility of using the melt and dilute process at SRS to treat sodium-bonded driver spent nuclear fuel was considered. See revised Section 2.6 of the EIS for a discussion on why this alternative was dismissed from further evaluation.

Comments from the North Augusta, South Carolina, Public Hearing August 17, 1999		
No.	***Comment***	***DOE Response***
Patricia McCracken		
503-1	"One of the things that really...struck me about this EIS...was that there seems to be a predecisional legal agreement that the DOE has made with Idaho, and that decision really preempts the EIS. And it really makes the DOE not have a national environmental policy, but rather is, in the case of Idaho, setting a precedent to look at a waste before you have the EIS or before there's some comment or where people have an opportunity to comment at all on it. So I think that's one of the things that this—this has really struck me as...being not a national policy. I hope I can get some more information on that case, and really that was a comment that should have been included in the EIS."	DOE is responsible for developing and maintaining a capability to safely manage its spent nuclear fuel. As stated in the introduction to the EIS, the SBSNF EIS follows the June 1995 Record of Decision (60 FR 28680) for DOE's Programmatic Spent Nuclear Fuel EIS, in which DOE decided to regionalize spent nuclear fuel management by fuel type for DOE-owned spent nuclear fuel. DOE also decided to: (1) continue environmental restoration activities at INEEL; (2) develop cost-effective treatment technologies for spent nuclear fuel and waste management; and (3) implement projects and facilities to prepare waste and treat spent nuclear fuel for interim storage and final disposition. This Record of Decision provides the programmatic umbrella for the site-specific actions addressed in the SBSNF EIS, as well as the Savannah River Spent Nuclear Fuel Management EIS and the Idaho High-Level Waste and Facilities Disposition EIS. The Savannah River Spent Nuclear Fuel Management EIS evaluates the impacts from the treatment of aluminum-clad and other spent nuclear fuel designated for treatment at SRS. The Idaho High-Level Waste Draft EIS evaluates the impacts from processing specific amounts of calcined high-level and sodium-bearing radioactive waste material currently located at INEEL. The materials (spent nuclear fuel and high-level radioactive waste) addressed in these EISs have unique characteristics and requirements which necessitate their separate evaluation. In a related action alluded to by the commentor, in a 1995 agreement with the State of Idaho (the Settlement Agreement and Consent Order issued on October 17, 1995), DOE committed to removing all spent nuclear fuel from Idaho by 2035. More than 98 percent of DOE's sodium-bonded spent nuclear fuel is located at INEEL and is subject to the requirements of this Settlement Agreement and Consent Order. Copies of the Settlement Agreement and Consent Order were made available to the public at the public meetings and are also located in the public reading rooms, and in Appendix K of the EIS.

Comments from the North Augusta, South Carolina, Public Hearing August 17, 1999		
No.	***Comment***	***DOE Response***
503-2	"I've commented, even in Nevada, mainly about how small business could be incorporated into some of these...various processes. And I did get a copy of the cost report and reviewed your references. We would certainly like in our area to get a volunteer group together and possibly make some phone calls along with the people that your agency is calling. I think your contractor and his procurement process has a very narrow group of people in which personal communication—I mean, I just felt like some of this was not documented real well and hope we can work with y'all [*sic*] later. And we have some small businesses that would certainly like to have a chance, whatever you decide to do, that we can also give you some of the cost here. If we could get some specifications which I think are lacking in the EIS, I have commented more on that. Who do we contact...in terms of maybe expanding your base of phone calls in terms of...I noticed you called the U.S. Tool and Die on their cost to fabricate C-22, some kind of pipe. Maybe we could do that too. You think we could call some of our people? Who would I contact at your agency so that we could get some volunteer calling going on in our area? We'd like to have some business here. When some of the people here say they think they can do some processes, I hope you'll look at that. I think they have given some excellent presentations at the meetings I've been to and I've been very impressed with them. I think I heard we can do it back here. So I hope y'all [*sic*] do look at some of the other technical issues."	Contacts with the businesses identified in the Cost Study were made to get estimates that were used for comparative purposes in the Cost Study. These contacts were not part of a procurement process.

Comments from the North Augusta, South Carolina, Public Hearing August 17, 1999		
No.	***Comment***	***DOE Response***
503-3	"I think your computer model is lacking in those numbers in terms of health effects because, without the technology, I don't know how you could decide what the numbers are in many cases."	The GENII computer program used to estimate the human health effects from releases of radioactive material during normal operation and accidental conditions is a well-known program, and its applicability has been demonstrated in various DOE EISs. The program models the dispersion of releases and calculates potential doses to the public and individuals residing in the vicinity of the facility. All required input to this program is well defined and the process is well understood. The evaluation is independent of the technology and equipment used. The only input from each process to this program is the quantity of radioactive material released during normal and accident conditions. As explained in the response to comment 16-47, the releases were estimated based on facility safety analysis reports. The atmospheric dispersion of radioactive material releases vary depending on the type and duration of the release. The selection of a dispersion model is an input to the GENII computer program. The dispersion models used in the program are well defined and are explained in Appendix E. These models are independent of the technologies used. The expression "new environmental equipment" is not used in the EIS and new environmental equipment is not related to the use of a computer program. Contamination in the off-gas system filters originates from the process. Each process is well defined. For example, because of the high temperature used in the melt and dilute process, some radionuclide elements with boiling temperatures below the process temperature would evaporate, while some elements would be oxidized and released to the off-gas system. The gaseous flow through the off-gas system first would be condensed and adsorbed, and then would be filtered before entering the atmosphere. All noble gases would pass through the filters, but only a small fraction of particulates would pass through the filters. The specific assumptions on various filtration factors are given in Appendix E and Appendix F. These appendices also provide the source terms associated with each of the releases considered.
503-4	"I've been very impressed with the EISs at Savannah River. And I've reviewed some of this and I hope I can continue to... give comment on this."	The commentor's statement concerning EISs at SRS is noted. DOE welcomes comments on all of its NEPA actions.

Comments from the North Augusta, South Carolina, Public Hearing August 17, 1999		
No.	***Comment***	***DOE Response***
503-5	"Well, I disagree totally with it by the way your computer models and how they.... I would...really like to look at how they got those numbers." *[Commentor refers to computer modeling of PUREX wastewater discharges]*	PUREX at SRS is the only treatment that would result in discharges of radionuclides or nonradioactive hazardous chemicals to surface water. The major sources of this liquid effluent would be process cooling water and steam condensate from the auxiliary facilities that support PUREX processing. As described in Section 4.5.2, the mechanism associated with releases of liquid effluent from PUREX processing is essentially independent of the type of fuel processed. The released quantities are the measured values provided in the SRS Site Environmental Report for 1997.
Ernest Chaput		
504-1	"I want to congratulate the Department for recognizing the need to develop a disposition strategy for this fuel which is intended to go to Yucca Mountain. We all hope Yucca Mountain comes out. I know this is a direct issue for the draft EIS on Friday and so that's — that's a very big step. We congratulate you for trying to recognize your responsibility, nuclear responsibility, to safely disposition the fuels that were left over now that the Cold War is won and other nuclear programs ...are being shut down and other programs are taking over the cleaning up that you've done. We believe, from...my understanding of the waste acceptance criteria of the draft, that some kind of a treatment will be mandatory, and so we commend you for doing that."	The commentor's expressed support for DOE's action to proceed with an EIS for the treatment and management of sodium-bonded spent nuclear fuel is noted. In accordance with the Nuclear Waste Policy Act of 1982, DOE is committed to the development of a licensed national repository for spent nuclear fuel and high-level radioactive waste and is engaged in activities to fulfill this commitment. A Yucca Mountain Draft EIS was issued by DOE in July 1999.
504-2	"...we notice that, as you pointed out, two of the six alternatives included in the draft include the shipment of the blanket materials to Savannah River for treatment either by the PUREX process or by the proposed melt and dilute facility. As a policy in my organization, we do not support the shipment of waste materials to Savannah River unless it can be clearly demonstrated that Savannah River has a significant capability or advantage to perform the task which cannot be reasonably established at the generated site. In other words, don't bring your waste to South Carolina unless you can clearly demonstrate you can't handle it somewhere else, particularly, preferably, the generating site."	The commentor's objections to the shipment of spent nuclear fuel to SRS for treatment is noted. The selection of reasonable alternatives evaluated in the EIS was made in accordance with Council on Environmental Quality Regulations (40 CFR 1500-1508) and DOE's NEPA-related regulations (10 CFR 1021) and procedures. In addition, as discussed in Section 1.3 of the EIS, the selection of reasonable alternatives was done in response to the issues raised during the public scoping period.

Comments from the North Augusta, South Carolina, Public Hearing August 17, 1999		
No.	*Comment*	*DOE Response*
504-3	"...if the Department determines that the shipment of blanket elements to the Savannah River Site is in the national interest, then we strongly recommend that only the PUREX treatment option be considered. Our reasons are twofold: One, PUREX is currently operational. The big concern, our big fear in South Carolina, is people ship us waste that eventually ends up being untreatable or it doesn't get treated at all and ends up resident in South Carolina. We want a clear path of any waste coming into the state, we want a path going out. And that path is the PUREX-DWPF-National Repository. The proposed melt and dilute facility is currently in development. The waste forms have not been extensively reviewed for acceptance in the national repository. The program is underfunded, potentially behind schedule. The inclusion of this material will further complicate its process development and facility operation. And...there is no assurance that the product form will be ultimately accepted into the National Repository and so, therefore, we...our strong recommendation is, if you do consider Savannah River, canyons is the only thing that my organization personally finds acceptable."	The commentor's preference for using PUREX processing instead of melt and dilute at SRS is noted. The final decision on the process to be selected for treating the sodium-bonded spent nuclear fuel will be based on the impacts provided in this EIS along with the conclusions presented in the Cost Study and Nonproliferation Impacts Assessment. The commentor is correct that the melt and dilute process at SRS is currently under development. However, based on recent research and development activities, preliminary conceptual design work, and technical maturity, DOE considers melt and dilute to be a viable technology option that can be implemented at SRS or ANL-W. DOE expects the waste generated from this process would meet the geological repository acceptance criteria.
504-4	"If it does come to the canyons, it has to come with adequate budgetary resources. We've got lots of other important missions on this site and we've got to make sure they...are carried on also. And so we would expect or require a firm DOE commitment for incremental funding....And if Savannah River capabilities are being considered, then only PUREX should be considered and then only if additional—adequate funding is provided."	If DOE selects Alternative 3 in the Record of Decision, use of the F-Canyon at SRS for blanket spent nuclear fuel treatment would not begin without the assurance of adequate funding. However, Congress determines how funds are allocated. DOE spends monies consistent with Congressional direction. DOE is not in a position to make the difficult tradeoffs that may be required between alternative Federal programs and spending priorities. The issue of funding for the treatment and management of sodium-bonded spent nuclear fuel is beyond the scope of the SBSNF EIS.
504-5	"The draft EIS identifies the electrometallurgical facility which currently exists at Argonne-West and...it initially appears...that [facility] can meet that criteria."	As discussed in Section 2.4.1 and 2.5.2 of the EIS, with a few equipment modifications, existing facilities at ANL-W would be suitable to accommodate the electrometallurgical treatment of sodium-bonded spent nuclear fuel.

Comments from the Boise, ID, Public Hearing *August 24, 1999*		
No.	*Comments*	*DOE Responses*
Steve Hopkins		
600-1	"I would like to see the comment period extended since the nonproliferation and cost reports have just been released."	In an effort to ensure that all interested parties had time to comment on the draft EIS, the deadline for transmittal of comments was extended from September 13 to September 28, 1999 (64 FR 49169). This extension also provided additional time for public review of the Cost Study and Nonproliferation Impacts Assessment. However, it should be noted that comments related to these reports are not within the scope of the EIS.
600-2	"Even though it is realized that these [nonproliferation and cost] reports are not part of the NEPA process, it is the only chance for the public to comment on them."	As noted by the commentor, although the Nonproliferation Impacts Assessment and Cost Study are not part of the NEPA process, the public may comment on them during the comment period for the draft EIS. In fact, DOE expedited the completion of these reports so that they would be available to the public to review in conjunction with the draft EIS. These reports were mailed to interested parties on August 12, 1999, and were made available to attendees at all of the public hearings on the draft EIS. DOE also extended the comment period from September 13 to September 28, 1999, (64 FR 49169) to provide the public with additional time to make comments.
600-3	"This is the public's only opportunity to comment, and you're starting an environmental impact statement process before having the final results [of the demonstration project] in. The demonstration project that you made, you have got enough already to do your draft EIS, but the public has to be taken into account in terms of it [how] should be completed before moving on with an EIS. The purpose was to demonstrate that it could work. It's called a demonstration project. And you're moving forward, analyzing an alternative that the public doesn't have any data [on] at this point in terms of the results."	The Electrometallurgical Treatment Research and Demonstration Project was successfully completed in August 1999, and the final results of the National Research Council's independent review of the project was published in April 2000. The commentor is correct in stating that DOE used the results of the demonstration project in preparing the draft EIS. Information available on the demonstration project includes the environmental assessment, published in 1996, as well as a series of independent status reports published by the National Research Council. This information was placed in the public reading rooms and, thus, was made available to the public.

Comments from the Boise, ID, Public Hearing August 24, 1999		
No.	*Comments*	*DOE Responses*
600-4	"I understand there's a second comment period after the Final [EIS] is issued with the preferred alternative. However, it's, like, 99 percent of the time or greater that when you have a preferred alternative that's what's [*sic*] the Record of Decision. So you can argue that you can have a public comment period, but the comments are not taken into consideration. Supposedly, in this process, you're factoring in the public's comments to make your preferred alternative, although you can argue you're not doing that at all."	Although the NEPA process does not provide a formal comment period with public hearings following publication of the final EIS, DOE welcomes comments. These comments can be made during the 30-day period between publication of the EIS and issuance of the Record of Decision. DOE considered all of the comments received during the public comment period on the draft EIS. Public comments are one of several factors considered in identifying a preferred alternative. The selection of a method for treating and managing DOE's sodium-bonded spent nuclear fuel will be published in the Record of Decision. Factors taken into consideration when making that decision include the analyses presented in the EIS, public comments, cost, schedule, technical assurance, policy, and program objectives.
600-5	"...at one point in this [Electrometallurgical treatment] process you're separating out highly enriched uranium. That's reprocessing. That may not be a final waste stream, but it's a reprocessing technology for separating out highly enriched uranium.... [*in response to a presenter's statement that the nonproliferation report concludes that electrometallurgical treatment is in compliance with all of the U.S. nonproliferation goals and policy*]...That's bunk. It's a reprocessing technology ....The Department of Energy has conveniently reworked the definition of reprocessing to fit the situation, so it's not technically reprocessing under the new definition. But under the definition of what reprocessing does, this is absolutely reprocessing."	The assessment of nonproliferation impacts is not part of the scope of the EIS. However, none of the alternatives analyzed in this EIS, except PUREX processing at SRS, would generate weapons-usable fissile materials. Although highly enriched uranium would be an interim product, it would be down-blended to low-enriched uranium during electrometallurgical treatment. Within the current equipment configuration and design, it is not possible to produce weapons-usable plutonium by adjusting operating parameters. Traditional aqueous processing would have to be used after electrometallurgical treatment (pyroprocessing). However, traditional aqueous processing could also be used to produce weapons-usable plutonium directly from the spent nuclear fuel, without pyroprocessing. The United States' policy on nonproliferation is contained in Presidential Decision Directive 13, a classified document. At the time the Presidential Directive was signed, an unclassified press release stated that, "The U.S. will seek to eliminate where possible the accumulation of stockpiles of highly-enriched uranium or plutonium." This would be done by down-blending the highly enriched uranium in the driver spent nuclear fuel and immobilizing the plutonium in the ceramic waste form. The press release also stated that the United States "does not itself engage in plutonium reprocessing for either nuclear power or nuclear explosive purposes."
600-6	[*in reference to the Nuclear Waste Policy Act*]: "That Act can be amended. Congress spent all of an hour on that before they went off on their vacation for Christmas. That's one of the most bogus acts that's ever come across the radar screen in this country."	The actions of elected officials are beyond the scope of this EIS.

Comments from the Boise, ID, Public Hearing *August 24, 1999*		
No.	*Comments*	*DOE Responses*
600-7	"...even though the Department is supposedly committed to building a repository, it's still very possible that a repository will not be open in the near future. I mean, at the earliest possible date, it would be open to accept spent fuel would be what—2010, 2012, something like that. That's 10 years away. And yet, there's lots of other spent fuel that could go directly to the repository where the Waste Acceptance Criteria are currently from INEEL. So, it's not like you're looking at the earliest possible date 10 years away that anything needs to be done with the spent fuel, especially when it's continually reasserted that it poses no significant environmental problem right now. You're only talking about a problem as it exists in a repository."	In accordance with the Nuclear Waste Policy Act of 1982, DOE is committed to the development of a licensed national repository for spent nuclear fuel and high-level radioactive waste and is engaged in activities to fulfill this commitment. As stated in the introduction to the EIS, the programmatic risk in implementing any of the potential alternatives for the treatment and management of sodium-bonded spent nuclear fuel, or of not treating this fuel, is the uncertainty surrounding the acceptability of DOE spent nuclear fuel for emplacement in a potential geologic repository. Although not final, the latest guidance provided by DOE's Office of Civilian Waste Management in their "Waste Acceptance System Requirements Document," Revision 3, April 1999 (see Section 4.12.1 of the EIS), indicates that it is highly probable that sodium-bonded spent nuclear fuel would not be acceptable in the repository without some stabilization and/or removal of the metallic sodium. The stabilization of the spent nuclear fuel and/or removal of the metallic sodium will provide greater protection of human health and the environment. Having completed the Electrometallurgical Treatment Research and Demonstration Project (see Section 1.6.3) and in planning the closure of its PUREX processing capabilities, DOE now needs to decide whether these processes are suitable for treating the remaining sodium-bonded spent nuclear fuel, or whether there is sufficient reason to delay a decision and wait for the development of other treatment technologies. Delaying the EIS could result in the loss of capability and of experienced, knowledgeable technical staff should DOE decide at a later date to use the electrometallurgical process to treat sodium-bonded spent nuclear fuel. Section 1.2 of the EIS has been revised for clarification.
600-8	"...[Electrometallurgical treatment] treatment [of the sodium-bonded spent nuclear fuel] ... may not be required. That's my main point. You don't know that it's going to be required."	The focus of this EIS is to assess the potential environmental and health impacts associated with the treatment and management of sodium-bonded spent nuclear fuel. See response to comment 600-7.
600-9	"...without that [NAS National Research Council Waste Characterization] report, it's hard for the public to know what's going to happen with all these different waste streams."	The expected fate of each waste stream is identified in the EIS. The National Academy of Sciences' National Research Council Committee assessment of waste form development and characterization is available in the DOE public reading rooms.

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600-10	"Because you're basically, without treatment, the spent fuel, you have got one form of waste even though it's not technically referred to as waste now by the Department of Energy. You do the processing and you have got various waste streams that have not been characterized yet. How is the public to react to that in terms of what we're going to do with this and that waste stream if they're not defined? If they're not defined, they don't have a destination."	All of the alternatives evaluated in this EIS would produce some forms of high-level radioactive waste. Electrometallurgical treatment (pyroprocessing) would produce two new waste forms, both of which are more stable than untreated sodium-bonded spent nuclear fuel. DOE expects that these waste forms would be suitable for disposal in a geologic repository.
600-11	"You don't seem to take [the National Research Council's report on DOE's claims concerning the Electrometallurgical treatment demonstration project] too seriously, but the public does, because I don't think the public has a whole lot of trust in Argonne, sorry to say. But the U.S. Nuclear Regulatory Commission is an independent body, and I'm not saying they have instantly more credibility. But that's important, that verification or nonverification, and we don't have that yet."	DOE commissioned the National Academy of Sciences' National Research Council review of the electrometallurgical treatment technology in 1995. Early Committee reports were instrumental in the DOE's redirection of the Argonne program to concentrate on demonstrating the technology for sodium-bonded metal fuel. DOE will consider the final National Research Council report in making a decision on how to proceed with the treatment and management of the sodium-bonded spent nuclear fuel.
600-12	"The other thing [is] we can't use [the Nuclear Regulatory Commission report on the Electrometallurgical treatment demonstration project] to comment until the final EIS is out and [it] doesn't do much to hear the comment at that point, because you basically take what the preferred alternative is in the final EIS, and that's your Record of Decision. So it's a formality at that point."	While the final National Research Council report on the Electrometallurgical Treatment Research and Demonstration Project was published in April 2000, the Council's interim status reports on the project were made available in the public reading rooms. Thus, prior to making comments on the draft EIS, the public had an opportunity to review all of the information that was made available by the National Research Council and was used to prepare the EIS. DOE will consider the data contained in the final National Research Council report in preparing the Record of Decision.

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No.	*Comments*	*DOE Responses*
600-13	"...to refer to this technology as not reprocessing is so dishonest, so disingenuous. This is absolutely a reprocessing technology. ...Hazel O'Leary actually said in 1994 that this technology is the essential processing technology for IFR. And I know that you're saying that it's been amended but, in essence, this technology was designed to separate out plutonium. And that plutonium, based upon our nonproliferation stance, ran contrary to our nonproliferation stance, so we essentially killed IFR on those grounds. And here we have the most proliferable dangerous aspect of IFR still alive. And that runs very contrary to what we were given in the early '90s, which we were taking some responsible steps to set an example for the rest of the world not to reprocess. It doesn't mean that there aren't countries that are reprocessing. But our intent was to discourage other countries from reprocessing, to take that step in order to acquire bomb grade material. And here, you have a reprocessing technology that's being used. I know this material, for instance, the highly-enriched uranium is not going to be used for bombs, but it is bomb material; therefore, it's a reprocessing technology. And you're keeping alive a reprocessing technology that's, from my point, more dangerous than PUREX, because it can be more easily concealed. You can put this technology underground, where PUREX would be very difficult to do. Quote from a previous NAS study, because there have been many, quote: 'Probably the greatest hazard arises from spreading sophisticated technologies around the world, technologies which make reprocessing spent fuel easier and possible in facilities small enough to conceal underground.' That's directly from the NAS related to this technology. To quote professor James Warf from the University of Southern California, Professor of Chemistry, Emeritus, 'with some modifications plutonium could be produced.' To quote an Argonne spokesperson at the site in 1995, 'We could easily modify the technology to produce plutonium.' Another NAS conclusion, quote: 'could be redirected to produce material with nuclear detonation capability.' That report also raised questions about the interim storage of the waste streams and other aspects of pyroprocessing.	As stated in the Nonproliferation Impacts Assessment, the alternatives involving PUREX reprocessing and broad application of electrometallurgical treatment of both driver and blanket fuel have a greater potential to provide encouragement to other countries to engage in plutonium reprocessing. Given the small quantity and unique characteristics of the sodium-bonded spent nuclear fuel and the reason for the treatment, however, such encouragement, if any, would be limited. In addition, electrometallurgical treatment (pyroprocessing) would not result in an increase in weapons-usable fissile material inventories. Although highly enriched uranium would be an interim product, it is would be down-blended to low-enriched uranium during electrometallurgical treatment. As stated in response to comment 600-5, within the current equipment configuration and design, it is not possible to produce weapons-usable plutonium by adjusting operating parameters. Traditional aqueous processing would have to be used after electrometallurgical treatment. However, traditional aqueous processing could also be used to produce weapons-usable plutonium directly from the spent nuclear fuel, without electrometallurgical treatment. The commentor also makes reference to the Integral Fast Reactor program. The purpose for the Integral Fast Reactor program was to develop an efficient, safe process for recycling nuclear fuel by using a liquid metal-cooled reactor in combination with an integral fuel reprocessing facility. As part of this program, the EBR-II was used for fuel-design and fuel irradiation testing. Congress canceled funding for the Integral Fast Reactor program in 1994.

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600-14	"It [the question of whether Electrometallurgical treatment should or should not be considered reprocessing and, therefore, proliferation-prone] kind of raises the question of exactly why you're proceeding with this technology at this point, which I have asked several times tonight, and I definitely have not gotten a reasonable response."	Although not final, the latest guidance provided by DOE's Office of Civilian Waste Management in their "Waste Acceptance System Requirements Document," Revision 3, April 1999 (see Section 4.12.1 of the EIS), indicates that it is highly probable that sodium-bonded spent nuclear fuel would not be acceptable in the repository without some stabilization and/or removal of the metallic sodium. The stabilization of the spent nuclear fuel and/or removal of the metallic sodium will provide greater protection of human health and the environment. Having completed the Electrometallurgical Treatment Research and Demonstration Project (see Section 1.6.3) and in planning the closure of its PUREX processing capabilities, DOE needs to decide whether this process is suitable for treating the remaining sodium-bonded spent nuclear fuel, or whether there is sufficient reason to delay a decision and wait for the development of other treatment technologies. Delaying the EIS could result in the loss of capability and of experienced, knowledgeable technical staff involved with the demonstration project should DOE decide at a later date to use the electrometallurgical process to treat sodium-bonded spent nuclear fuel. Section 1.2 of the EIS has been revised for clarification. DOE also conducted four independent nonproliferation assessments of the electrometallurgical treatment technology over the last 11 years. These assessments found the electrometallurgical treatment technology to be in accordance with U.S. nuclear nonproliferation policy for this specific application, and concluded that electrometallurgical treatment is not capable of separating plutonium in a form that would be suitable for weapons production.
600-15	"A DOE source was quoted in a trade journal...saying, quote: 'Just about the only thing they have left to do,' meaning Argonne, 'is this procedure.' And quote: 'it's a jobs issue.' That's what the DOE source said directly about this procedure. It's corporate welfare. This project has been featured twice on *The Fleecing of America*. I don't know of any other thing that's ever been featured twice. That's very significant. That never happens."	Congress determines how funds are allocated. DOE spends monies consistent with Congressional direction. DOE is not in a position to make the difficult tradeoffs that may be required between alternative Federal programs and spending priorities. The issue of funding for the treatment and management of sodium-bonded spent nuclear fuel is beyond the scope of the SBSNF EIS.
600-16	"From what I understand, too, the reactor has not even been completely drained of the spent fuel, which the money that's been going all along, $20 million a year since 1994, part of that was supposed to have gone towards draining the reactor. And from what I understand, that's not even done at this point."	The commentor's reference to the draining of sodium from the EBR-II reactor is not related to the subject matter of this EIS, which is the treatment and management of sodium-bonded spent nuclear fuel. The sodium-bonded spent nuclear fuel that is the subject of this EIS was removed from the EBR-II reactor and is currently stored at the Radioactive Scrap and Waste Facility at ANL-W.

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600-17	"Another NAS quote: 'Although developers of the electrometallurgical technique argue that the technology is proliferation-resistant, any spent fuel processing approach that's capable of separating fissionable materials from associated fission products and transuranic elements could be redirected to produce material with nuclear detonation capability. Demonstration of the process could, however, add to the risk that a nation intent on weapons production might consider adapting this technology for possible production of fissile material, although such material would be of poor quality for a weapon.' And that's disputable."	The Nonproliferation Impacts Assessment is not part of the scope of the EIS. Electrometallurgical treatment technology is not capable of separating weapons-usable plutonium. Traditional aqueous processing would have to be used after electrometallurgical treatment to produce weapons-usable material. However, traditional aqueous processing could also be used to produce weapons-usable plutonium directly from the spent nuclear fuel.
600-18	"I guess you just want to give money to Argonne. If that's the issue, then I'd just as soon that you not pursue reprocessing as the technology that's used."	The commentor's opposition to electrometallurgical treatment (pyroprocessing) is noted. The issue of spending money for electrometallurgical treatment is beyond the scope of the EIS.
600-19	"At this point, I have to support the No Action alternative, because it's the most reasonable alternative. There's no facility to accept waste. The Waste Acceptance Criteria are not finally known. The waste doesn't present any environmental threat due to the presence of sodium at this point. Obviously, spent fuel is dangerous. That spent fuel without sodium is still dangerous. So there's no clear justification for going forth with this technology at this point. So I support the No Action alternative."	The commentor's support for the No Action Alternative is noted. The EIS discusses the status of the waste acceptance criteria in Section 2.7 and the environmental impacts of the No Action Alternative in Section 4.2. The timing of DOE's decision on the treatment and management of sodium-bonded spent nuclear fuel in relation to the availability of a geologic repository is discussed in Section 4.12.2.

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No.	*Comments*	*DOE Responses*
Anonymous		
700-1	"We haven't appropriated the money for the [SRS melt and dilute] facility, and our cost study is based on that facility being operational when we compare disposal method. That looks like, to me, it's flawed."	DOE assumes that the SRS melt and dilute facility will be available to process blanket spent nuclear fuel in 2022. Many of the costs associated with this alternative, such as those for preparing and packaging the fuel for shipment to SRS, occur at ANL-W. Congress appropriates funds for the treatment of spent nuclear fuel. DOE spends monies consistent with Congressional direction. The issue of funding for the treatment and management of sodium-bonded spent nuclear fuel is beyond the scope of the SBSNF EIS. DOE believes that the Cost Study is adequate for the purpose intended. The results of the Cost Study will be among the factors considered during the decision-making process leading to the Record of Decision.
700-2	"The driver fuel, of course, is the one that's not usable in terms of the PUREX process because of the infiltrated sodium. So the candidates for taking care of the sodium there really lend themselves to...the electrometallurgical process. But that's only three metric tons.... But the big part of the project really is 57 metric tons of depleted uranium, in which plutonium is inbred. The sodium is removable from the surface of the uranium rods. And we [ANL-W/INEEL] have done that process mechanically and chemically a number of times to the tune of probably several thousand fuel rods. And they were, in fact, shipped to Atomic International, and then to Savannah River. The technology worked. It's very cheap. It's very gross. ...Where is it going to go? It's going to go someplace. It has to be removed if it's sodium. ...Why do we consider anything else, in terms of the blanket rods, because it has been done many, many times before at Argonne-West, and at Atomic International and at Savannah River?"	DOE agrees with the commentor that decladding and removal of sodium from blanket spent fuel have been performed many times in the past. Section 2.3.9 and Appendix C of the EIS describe the processes used in the past. As described in Section 2.5.3, DOE evaluated an alternative in which the cleaned (metallic sodium removed) blanket spent nuclear fuel would be packaged in high-integrity cans for storage and disposal in a geologic repository. In addition, DOE evaluated other alternatives where the cleaned blanket fuel would be treated further. The selection of various alternatives is a required step in performing an EIS that is in compliance with NEPA and Council of Environmental Quality regulations.
700-3	"Unless there's an incentive to reclaim or separate the plutonium from the depleted uranium rods, it makes absolutely no sense to me to do anything more than remove the cladding, remove the sodium, and store those rods, store those slugs, at Savannah River, or wherever they are in storage, much like spent fuel is stored. To... downgrade, or to whatever, just increases the proliferation problem."	The commentor's recommendation to remove sodium and place blanket spent nuclear fuel in cans is noted and is discussed in Section 2.5.3.

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No.	*Comments*	*DOE Responses*
John Commander		
701-1	"We support the treatment of the sodium-bonded spent nuclear fuel by the electrometallurgical process. The process should be used for all the fuel as described in Alternative 1 of the Draft Environmental Impact Statement. The electrometallurgical treatment has been proven to be satisfactory. Many of the other alternatives are in the concept or research stage. Nearly all of the sodium-bonded fuel is now at Argonne National Laboratory-West. It makes both common and economic sense to do the entire treatment there. ...Again, we support Alternative 1 very strongly."	The commentor's support for the electrometallurgical treatment of both driver and blanket sodium-bonded spent nuclear fuel (Alternative 1) is noted.
701-2	"I'm also concerned about the loss of jobs and skills if the treatment is not done at Argonne National Laboratory-West. These skills are particularly important at this time. The current administration is finally putting some new funding into the research—nuclear research and technology. And DOE has designated the INEL [*sic*] as a lead laboratory for this effort. We want to keep these qualified people here."	The commentor's concern that jobs and skills will be lost if treatment of sodium-bonded spent nuclear fuel is not conducted at ANL-W is noted. DOE recognizes the value and the presence of important skills at ANL-W and INEEL. As part of the decision-making process, DOE will consider the consequences of potential impacts to various environmental resources, including socioeconomics. The Record of Decision will explain the rationale and factors for DOE's decision.
701-3	"The electrometallurgical treatment has little risk that nuclear material could be diverted to use in nuclear bombs. The Draft-EIS has adequately answered the comments of those concerned about that risk."	The commentor is correct. Electrometallurgical treatment of sodium-bonded spent nuclear fuel would not produce weapons-usable material, thereby reducing the risk that this spent nuclear fuel might be diverted for other uses.
701-4	"Whatever alternative is chosen, it must meet the terms of the 1995 Governor's Agreement on Nuclear Waste. If treatment is done at the Savannah River [site], material must be moved there before the year 2035. And it is not clear to me that those facilities will be available to do any treatment before that year. This date is the deadline for all spent fuel to be out of Idaho."	Section 4.12.2 of the EIS presents a discussion on schedule consideration for the treatment and management of sodium-bonded spent nuclear fuel for each of the alternatives considered in the EIS. According to these schedules, the treatment of sodium-bonded spent nuclear fuel could be completed by 2035 for all treatment alternatives, including the direct disposal option of the No Action Alternative. Under the continued storage option of the No Action Alternative, the sodium-bonded spent nuclear fuel would be transferred out of the State of Idaho before the 2035 deadline. The availability of the SRS facilities for treatment of blanket spent nuclear fuel is also discussed in Section 4.12.2.

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No.	*Comments*	*DOE Responses*
Dick Kenney		
702-1	"I think that your calculation of background radiation of 360 millirems per year is considerably less than what the residents of Idaho Falls receive. I think you've left out several elements...in that calculation."	As shown in Table 3–8 of the EIS, the approximately 360 millirem per year natural background radiation dose is the sum of the calculated effective dose equivalent from terrestrial and cosmic sources (external dose) specific to the Snake River Plain area, as well as the estimated doses from cosmogenic sources and radon gas (internal dose) provided in the National Council on Radiation Protection and Measurements Report No. 93, which lists the average dose to an American. An individual in the Idaho Falls area may or may not receive this dose because of variations between geographic areas. The EIS provides a summary of various contributing sources of radiation in the vicinity of the INEEL site.
702-2	"Coalition 21 strongly supports the treatment of sodium-bonded spent fuel by the electrometallurgical process. The process should be used for both the driver and the blanket fuel, as described in Alternative No. 1. The ANL-West is...has successfully demonstrated that the electrometallurgical treatment works. We see no reason for additional research in other technologies. Let's do it, get the job done and be done with it."	The commentor's support for using the electrometallurgical treatment process to treat driver and blanket sodium-bonded spent nuclear fuel (Alternative 1) is noted.
702-3	"This alternative [Alternative 1], properly done, will make the remnants of the IFR program ready for final disposal. It will be done in a timely manner by a technology that is compatible with the IFR concept, we do not want sodium-bonded fuel still in storage. We do not want that fuel to be used as an example of another failed technology. This position is consistent with the objectives of our lawsuit against the Department of Energy regarding the IFR."	The commentor's support for Alternative 1, the electrometallurgical treatment of sodium-bonded spent nuclear fuel at ANL-W, is noted. The commentor makes reference to the Integral Fast Reactor program. The purpose for the Integral Fast Reactor program was to develop an efficient, safe process for recycling nuclear fuel by using a liquid metal-cooled reactor in combination with an integral fuel reprocessing facility. As part of this program, the EBR-II was used for fuel-design and fuel irradiation testing. Congress canceled funding for the Integral Fast Reactor program in 1994. The commentor's concern that the sodium-bonded spent nuclear fuel could be used as an example of "another failed technology" and whether DOE decides to retrieve or revive the Integral Fast Reactor concept is beyond the scope of this EIS. In the lawsuit referred to by the commentor ("Coalition 21 v. U.S. Department of Energy and Tammy L. Hobbes," Civil Case No. CV 98-0299-B-BLW), Coalition 21 seeks to require DOE to prepare an EIS to address the shutdown of the EBR-II and claims that DOE failed to examine the potential environmental consequences of this action. Since deactivation of EBR-II does not involve the treatment and management of sodium-bonded spent nuclear fuel, the objectives referred to by the commentor are beyond the scope of this EIS.

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No.	*Comments*	*DOE Responses*
702-4	DOE does not plan to generate more sodium-bonded fuel; thus, it is a limited program, one that can be solved and should be solved sooner, rather than later."	The commentor's support for the proposed action, the treatment and management of sodium-bonded spent nuclear fuel, is noted. As the commentor noted, with the shutdown and removal of all fuel from the EBR-II, DOE can no longer generate any additional sodium-bonded spent nuclear fuel at INEEL. Ninety-eight percent of the DOE-owned sodium-bonded fuel is now at the ANL-West and INTEC. Having completed the Electrometallurgical Treatment Research and Demonstration Project (see Section 1.6.3) and in planning the closure of its PUREX processing capabilities, DOE now needs to decide whether these processes are suitable for treating the remaining sodium-bonded spent nuclear fuel, or whether there is sufficient reason to delay a decision and wait for the development of other treatment technologies. Delaying the EIS could result in the loss of capability and of experienced, knowledgeable technical staff should DOE decide at a later date to use the electrometallurgical process to treat sodium-bonded spent nuclear fuel. Section 1.2 of the EIS has been revised for clarification.
Ted Carpenter		
703-1	"The tribes are renowned for use of resources efficiently and maximally. I support the electrometallurgical process because it does produce a separated uranium metal product. Once the earth has been invaded and the crust has been broken up to remove the rocks and the metal's been refined, let's keep using it, instead of considering it waste. The same thing goes for the fact that it separates out the stainless steel and noble metals—zirconium, niobium, nickel, chromium—all of those things. Those are resources; they are not waste."	Most of the noble metal fission products (e.g., niobium, technetium, ruthenium, rubidium, silver, cadmium, and zirconium) and fuel alloy (zirconium) in the electrorefiners would remain with the fuel cladding hull in the anode basket. In addition, some actinides would also remain with the noble fission products. The amount of material retained in the anode basket would strictly depend on the electrorefining operation conditions. If more actinides and the fuel matrix were dissolved in the molten salts, the retention of noble fission products would be lowered. The metal remains in the anode basket would be radioactive, and would be classified as high-level radioactive waste. It is true that electrometallurgical treatment has been used to produce metals from impure feedstock. However, that impure feedstock included metals with chemical contamination, not radioactive isotopes of the same metals. Noble metal recovery from the metallic waste would have limited uses because the metal would still be radioactive, (i.e., it would contain radioactive isotopes of the metal elements) and would still be considered radioactive metallic waste. However, uranium would be separated and could be used for other purposes. The disposition of this uranium, along with DOE's inventory of surplus uranium, will be determined through another NEPA review.

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No.	*Comments*	*DOE Responses*
703-2	"Also, of course, the fact that this Alternative 1 has minimal transportation across reservations simply avoids the issues of some of...the members who have fears."	As explained in the EIS, the risks associated with the fuel transport are very small. Regardless of the alternative, DOE would need to transport spent nuclear fuel and/or high-level waste out of the INEEL site. DOE will proceed in accordance with the DOE/Shoshone-Bannock Tribes Agreement-in-Principal, which covers notification and coordination of the transport of radioactive materials across the Fort Hall Reservation. Risks, including transportation, have been addressed in the EIS and will be considered by DOE prior to making any decisions regarding the treatment and management of sodium-bonded spent nuclear fuel.
Steve Herring		
704-1	"The options for the driver are really driven by the amount of sodium that is contained in the pores within the fuel. And, consequently, the electrometallurgic process is about the only viable alternative for getting that sodium out. For the blanket, it seems to me that we have a viable choice based on how well we can characterize the long-term longevity of those high-integrity cans. I understand that specifications can be written for them. But, if we write those specifications, that they have to be shown to be integral for 10,000 years, then we have a major testing program ahead of us for that. ...therefore,...if that is a driver on the cost of the options, then the electrometallurgical process should be used for the blanket, as well. However, if that is not a driver on the cost, then the use of high-integrity cans for the blanket assembly should be used for both of those, both Options 1 and 2, minimizing the amount of transportation.... And so, therefore, I would like to speak in favor of either Options 1 or 2."	The commentor's support for the use of the electrometallurgical process to treat driver sodium-bonded spent nuclear fuel is noted. The EIS does not present a cost comparison of the alternatives. However, a separate DOE Cost Study does compare the costs of each alternative. This Cost Study assumes that isolation of the treated spent nuclear fuel in a 10,000-year repository would rely on the integrity of other containment barriers rather than high-integrity can packaging.

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No.	*Comments*	*DOE Responses*
John Tanner		
705-1	"The treatment of the driver portion of the sodium-bonded nuclear fuel by the electrometallurgical process is the most sensible option proposed for the following reasons: It would allow recovery and use of the high-enriched uranium, which is valuable material that was costly to produce. This [driver] fuel is not suitable for the PUREX process, as already explained in the DEIS. The other methods, melt and dilute, chloride volatility, plasma arc ceramic, and so forth, are less well developed, are likely to be more expensive even after development, and involve heating the fuel to high temperatures, which will worry some people about whether the volatile elements would pollute the air."	The commentor's support for the treatment of driver sodium-bonded spent nuclear fuel by the electrometallurgical treatment method is noted. The EIS discusses all of the commentor's areas of concern. Separate studies consider the nonproliferation characteristics of the various alternative technologies and the costs associated with each of the alternatives. The EIS assessment and the conclusions presented in the separate studies will be considered during DOE's decision-making process, the results of which will be published in the Record of Decision.
705-2	"The plutonium in the blanket fuel is also valuable and should be recovered. If this [plutonium recovery from the blanket fuel] were done by the PUREX process, the recovered plutonium would be pure enough to be made into mixed oxide fuel to generate electricity in commercial power reactors. Much of the development of this [PUREX] process is already contemplated for plutonium recovered from weapons. The cost of decladding, sodium removal, and shipment from Idaho would, of course, need to be considered. The plutonium could also be recovered by the electrometallurgical process. Why is this not mentioned as an alternative in the DEIS? This is as reasonable as many of the other alternatives presented. Although the recovered plutonium would be too contaminated with other transuranic elements to be useful as MOX fuel, it would be useful in a future fast neutron reactor, such as the one which produced it.	The commentor's remarks about the value of plutonium present in the sodium-bonded spent nuclear fuel are noted. The intent of this EIS, as discussed in Section 1.2, is to resolve issues associated with the sodium content of sodium-bonded spent nuclear fuel. The disposition of the fissile material content of the fuel is not within the scope of the EIS and is not considered an issue in the formulation of the reasonable alternatives. It is, however, an important consideration in the Nonproliferation Impacts Assessment of the alternatives that was prepared separately from the EIS. The conclusions of the Nonproliferation Impacts Assessment, along with those of the EIS, will be considered during the decision-making process leading to the Record of Decision.

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No.	*Comments*	*DOE Responses*
705-3	"But to answer the question just raised, recovery of plutonium by the electrometallurgical process was omitted in order to please influential antinuclear critics who raised weapons proliferation concerns, ignoring the fact that the electrometallurgical process is far more proliferation- resistant than the well-known PUREX process. The demonstration of plutonium separation by the electrometallurgical process would do nothing to aid anyone's ability to obtain weapons-usable material." "However, putting this plutonium in the waste, as proposed for most of the alternatives in the DEIS, will only temporarily please these critics. When it is later proposed to bury this waste, whether in Yucca Mountain or elsewhere, they will again object, pointing to plutonium's long half-life and to recent evidence that trace amounts of plutonium can migrate in groundwater under special artificial conditions. Note that the critics have been vehemently opposing the transport and burial of waste with only trace amounts of plutonium in the WIPP. What will they say when it is proposed to bury waste with substantial amounts of plutonium?"	DOE, consistent with U.S. nuclear nonproliferation policy, would not separate plutonium except for the PUREX process. DOE expects that the plutonium-containing waste from the electrometallurgical treatment process would be acceptable in a geologic repository for the same reasons that plutonium-containing commercial spent nuclear fuel is already acceptable.
705-4	Any method of dealing with plutonium will be criticized. Therefore, we should do the sensible thing and recover it for later use."	The commentor's remarks about the value of plutonium present in the sodium-bonded spent nuclear fuel are noted. The intent of this EIS, as discussed in Section 1.2, is to resolve issues associated with the sodium content of sodium-bonded spent nuclear fuel. The disposition of the fissile material content of the fuel is not within the scope of the EIS and is not considered an issue in the formulation of the reasonable alternatives. It is, however, an important consideration in the Nonproliferation Impacts Assessment of the alternatives that DOE prepared separately from this EIS. The conclusions of the Nonproliferation Impacts Assessment and those of the EIS will be considered during the decision-making process.
Beatrice Brailsford		
706-1	"I think you have done a good job in the draft EIS, demonstrating that nothing needs to be done with the blanket fuel, as far as for the processing beyond the removal of the sodium in mechanical ways in which we know how to do...certainly for the blanket, no action is the appropriate course."	The commentor's opinion that the appropriate course for blanket sodium-bonded spent nuclear fuel is sodium removal and direct disposal (Alternative 2 for blanket fuel), is noted.

Comments from the Idaho Falls, Idaho, Public Hearing *August 26, 1999*		
No.	*Comments*	*DOE Responses*
706-2	"As you know, we have asked for an extension of this comment period. ...And it seems to me that...you really are looking at a real rush job to try to finish this up by the end of the year. So, I would encourage you to extend the comment period on the draft EIS..."	In an effort to ensure that all interested parties had time to comment on the draft EIS, the deadline for transmittal of comments was extended from September 13 to September 28, 1999 (64 FR 49169). DOE did not rush the preparation of the EIS. By extending the comment period, it provided the public with additional time to consider and make comments on the document.
706-3	"[Extend the comment period] ...at least until the NRC [National Research Council] analysis comes out. I received the cost study and the nonproliferation report today. And I won't receive the NRC report until December, simply because you won't either."	While the final National Research Council report on the Electrometallurgical Treatment Research and Demonstration Project was published in April 2000, the Council's interim status reports on the project were made available in the public reading rooms. Thus, prior to making comments on the draft EIS, the public had an opportunity to review all of the information that was made available by the National Research Council and was used to prepare the EIS. DOE will consider the data contained in the final National Research Council report in preparing the Record of Decision.

Comments from the Arlington, Virginia, Public Hearing August 31, 1999		
No.	*Comments*	*DOE Responses*
Maureen Eldredge		
800-1	"I take offense at talking about nuclear processes and telling the public that it's like common table salt—that you can go buy it in the grocery store. It's just an aside that I urge you not to use that kind of language."	The commentor is referring to an analogy used in the DOE presentation on August 31, 1999, to explain the disposition of metallic sodium in the sodium-bonded spent nuclear fuel during electrometallurgical treatment. As stated in the EIS, during electrometallurgical treatment the metallic sodium would be converted into a nonreactive form (sodium chloride) and would be disposed of with the high-level ceramic radioactive waste product. In the DOE presentation, the nonreactive sodium chloride form was described as analogous to "common table salt." It was not DOE's intent to mislead the public to believe that they could buy this "salt" in a grocery store; rather, DOE sought to communicate to the public what happens to the metallic sodium during treatment.
800-2	"...you mentioned the need to make a decision regarding PUREX because the [SRS] canyons will be shutting down. Do you have a schedule for that shutdown? I was not aware there was an actual date certain."	The plans for shutdown are being developed. Therefore, if PUREX processing were selected, sodium-bonded blanket fuel would need to be placed on the schedule.

Comments from the Arlington, Virginia, Public Hearing August 31, 1999		
No.	*Comments*	*DOE Responses*
800-3	"This project and the need for it in terms of the repository is completely incompatible with the schedule that Yucca Mountain is on. Not only are the waste criteria not set, there are growing concerns about the feasibility of that site as a repository and at least five years out, if not longer, before those kinds of decisions would be made." "I think probably you could add to a list of 'why now', the Federal Budget process with the Fiscal year 2000 starting on October 1st and the problem this project ran into in that they wouldn't be able to justify spending money if suddenly they weren't going to have a ROD into the middle of the fiscal year. Perhaps I'm just being cynical."	In accordance with the Nuclear Waste Policy Act of 1982, DOE is committed to the development of a licensed national repository for spent nuclear fuel and high-level radioactive waste and is engaged in activities to fulfill this commitment. This commitment is ongoing. The EIS does not assume that Yucca Mountain will be selected as the high-level waste repository. It only assumes that, at some time in the future, a geologic waste repository will be licensed and operated by DOE which will receive spent nuclear fuel and high-level radioactive waste. As stated in the introduction to the EIS, the programmatic risk in implementing any of the potential alternatives for the treatment and management of sodium-bonded spent nuclear fuel, or of not treating this fuel, is the uncertainty surrounding the acceptability of DOE spent nuclear fuel for emplacement in a potential geologic repository. Although not final, the latest guidance provided by DOE's Office of Civilian Waste Management in their "Waste Acceptance System Requirements Document," Revision 3, April 1999 (see Section 4.12.1 of the EIS), indicates that it is highly probable that sodium-bonded spent nuclear fuel would not be acceptable in the repository without some stabilization and/or removal of the metallic sodium. The stabilization of the spent nuclear fuel and/or removal of the metallic sodium will provide greater protection of human health and the environment. Having completed the Electrometallurgical Treatment Research and Demonstration Project (see Section 1.6.3) and in planning the closure of its PUREX processing capabilities, DOE needs to decide whether these processes are suitable for treating the remaining sodium-bonded spent nuclear fuel, or whether there is sufficient reason to delay a decision and wait for the development of other treatment technologies. Delaying the EIS could result in the loss of capability and of experienced, knowledgeable technical staff should DOE decide at a later date to use the electrometallurgical process to treat sodium-bonded spent nuclear fuel. Section 1.2 of the EIS has been revised for clarification.
800-4	"I believe that the whole point of looking at cumulative impacts was that you might have a series of nonsignificant impacts which, when added up would become an impact. So I urge you to look at that again."	As described in Section 4.11 of the EIS, cumulative impacts are defined by the Council on Environmental Quality as the environmental impacts that result from the incremental impact of the action when added to other, past, present, and reasonably foreseeable future actions, regardless of what agency (Federal or nonfederal) or person undertakes such other actions. This section provides the discussion on the cumulative impacts for all resources evaluated in the EIS. For each resource, where the incremental impact from an action would be very small, its contribution to the cumulative impacts would be insignificant.

Comments from the Arlington, Virginia, Public Hearing August 31, 1999		
No.	*Comments*	*DOE Responses*
800-5	"Once again, as always, we do not believe there is a need for this action. It's our continued belief that this project is not proceeding because of any need, but rather the political need to retain jobs at Argonne West, retain missions, and leave the door open for their future dream of getting more waste forms to process. That hope has been revitalized in many ways, including Senator Domenici's attempts to start a new Office of Reprocessing. So I think it's a realistic hope on their part and one of the reasons we are continuing to oppose this project."	DOE is responsible for developing and maintaining a capability to safely manage its spent nuclear fuel. To ensure that the State of Idaho Settlement Agreement and Consent Order is met, and to facilitate disposal, DOE needs to reduce the uncertainties associated with qualifying the sodium-bonded spent nuclear fuel for disposal. While DOE notes the commentor's belief that the need for the proposed action is concerned with the political need to retain jobs and missions at ANL-W and the hope of having more waste forms to process, this comment is beyond the scope of this EIS. See response to comment 800-3.
Hisham Zerriffi		
801-1	"...the major purpose of this action is to remove the reactive sodium, toxic-sodium from the spent fuel. Now, for most of the alternatives...or some of the alternatives at least, for the blanket spent fuel you are going to do that removal process at Argonne using the process described in Section 2.4.9, which is a fairly simple process, it seems. And then run it through PUREX? What's the point of the second part of that, exactly, if you've already removed the sodium in the Argonne hot cell?"	The programmatic risk in implementing any of the potential alternatives for the treatment and management of sodium-bonded spent nuclear fuel, or of not treating this fuel, is the uncertainty surrounding the acceptability of DOE's spent nuclear fuel for emplacement in a potential geologic repository. While DOE has drafted preliminary waste acceptance criteria for a geologic repository, the final acceptance criteria will be more refined. If the repository is developed, final acceptance criteria will not be available until after the U.S. Nuclear Regulatory Commission issues its construction authorization based on successful demonstration of the safe, long-term performance of the repository in accordance with the U.S. Nuclear Regulatory Commission regulations. The presence of metallic sodium is the primary but not the only reason for the proposed action. The presence of metallic uranium or highly enriched uranium, could also complicate the process of certifying the repository. Such certification would require sufficient data and predictive analyses to demonstrate that emplacement of the spent nuclear fuel would not adversely affect a repository's ability to protect the environment and worker and public health and safety. To ensure that requirements of the State of Idaho Settlement Agreement and Consent Order are met and to facilitate disposal, DOE needs to reduce the uncertainties associated with qualifying sodium-bonded spent nuclear fuel for disposal. Appropriate treatment and management of the sodium-bonded spent nuclear fuel (e.g., PUREX processing) would significantly reduce complications related to disposal qualifications. The borosilicate glass waste form resulting from PUREX processing has been extensively tested and analyzed under conditions relevant to a geologic repository. DOE expects that other waste forms (e.g., ceramic and metallic) would be suitable for repository disposal.

Comments from the Arlington, Virginia, Public Hearing August 31, 1999		
No.	*Comments*	*DOE Responses*
801-2	"...I think the final EIS does need to clarify—yes, I understand that you have metallic uranium in the fuel and that is also an issue for the repository, as is the HEU. And I think that's not -- I mean, it's clear to me when I read through it but I think most of the public reading through it is not going to be very clear on that. That this is an issue of both sodium and the other metals and the HEU, and what of each of these are going to handle which part of that process? And I think that needs to be much more defined in the final EIS if you're going to do it."	Section 2.2 of the EIS states that the 60 metric tons of heavy metal of sodium-bonded spent nuclear fuel constitutes approximately 2 percent of DOE's total current spent nuclear fuel inventory of nearly 2,500 metric tons of heavy metal. According to the latest guidance provided by DOE's Office of Civilian Waste Management in their "Waste Acceptance System Requirements Document," Revision 3, April 1999, DOE spent nuclear fuel "may be accepted as bare fuel. The specific acceptance criteria for this bare fuel will be developed on a case by case basis." The decision, therefore, whether or not to treat spent nuclear fuel, including the N-Reactor fuel, before placement in a geologic repository has not been made. As discussed in Section 1.2 of the EIS, the presence of metallic sodium is the primary but not the only reason for the proposed action. The presence of metallic uranium, or the presence of highly enriched uranium, could also complicate the process of certifying the geologic repository. Such certification would require sufficient data and predictive analyses to demonstrate that placement of the spent nuclear fuel would not adversely affect a repository's ability to protect the environment and worker and public health and safety. To ensure that the State of Idaho Settlement Agreement and Consent Order is met, and to facilitate disposal, DOE needs to reduce the uncertainties associated with qualifying sodium-bonded spent nuclear fuel for disposal. Appropriate treatment and management of the sodium-bonded spent nuclear fuel would significantly reduce complications related to disposal qualifications.

Comments from the Arlington, Virginia, Public Hearing *August 31, 1999*		
No.	*Comments*	*DOE Responses*
801-3	"The IEER [Institute for Energy and Environmental Research] is disappointed that the Department has again issued a draft EIS which seems to sacrifice some pretty important environmental and nonproliferation goals to meet some programmatic goals which are questionable."	DOE is responsible for developing and maintaining a capability to safely manage its spent nuclear fuel. As stated in the introduction to the EIS, this EIS follows the June 1995 Record of Decision (60 FR 28680) for the "Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement," in which DOE decided to regionalize spent nuclear fuel management by fuel type for DOE-owned spent nuclear fuel. DOE also decided to: (1) continue environmental restoration activities at INEEL; (2) develop cost-effective treatment technologies for spent nuclear fuel and waste management; and (3) implement projects and facilities to prepare waste and treat spent nuclear fuel for interim storage and final disposition. The Record of Decision provides the programmatic umbrella for the site-specific actions addressed in this EIS, as well as the Savannah River Spent Nuclear Fuel Management EIS and the Idaho High-Level Waste and Facilities Disposition EIS. DOE is committed to improving its environmental management practices; to operating its facilities in a manner that meets or exceeds all applicable environmental, safety, and health requirements; and to cleaning up its environmental problems. The focus of this EIS is to assess the potential environmental and health impacts associated with the treatment and management of sodium-bonded spent nuclear fuel. Although not final, the latest guidance provided by DOE's Office of Civilian Waste Management in their "Waste Acceptance System Requirements Document," Revision 3, April 1999 (see Section 4.12.1 of the EIS), indicates that it is highly probable that sodium-bonded spent nuclear fuel would not be acceptable in the repository without some stabilization and/or removal of the metallic sodium. Stabilization of the spent nuclear fuel and/or removal of the metallic sodium would provide greater protection of human health and the environment. Having completed the Electrometallurgical Treatment Research and Demonstration Project (see Section 1.6.3) and in planning the closure of its PUREX processing capabilities, DOE now needs to decide whether these processes are suitable for treating the remaining sodium-bonded spent nuclear fuel, or whether there is sufficient reason to delay a decision and wait for the development of other treatment technologies. Delaying the EIS could result in a loss of capability and of experienced and knowledgeable technical staff should DOE decide at a later date to use the electrometallurgical process to treat sodium-bonded spent nuclear fuel. Section 1.2 of the EIS has been revised for clarification. DOE has also conducted an independent Nonproliferation Impacts Assessment of the treatment technologies analyzed in the this EIS. The Nonproliferation Impacts Assessment found all of the treatment technologies, except for PUREX processing at SRS, to be in accordance with U.S. nuclear nonproliferation policy.

Comments from the Arlington, Virginia, Public Hearing *August 31, 1999*		
No.	*Comments*	*DOE Responses*
801-4	"...there are no immediate health, environmental, and safety risks that need to be addressed immediately or that cannot be addressed through some sort of simple minimal preparation and fuel storage. I believe that's basically what the draft EIS states."	The timing for this action is a programmatic issue rather than a safety issue. As stated in Section 1.2 of the EIS, DOE considers it prudent to evaluate the alternative technologies now while DOE is performing site characterization activities for the potential repository at Yucca Mountain. See response to comment 801-3.
801-5	"There's no guarantee that Yucca Mountain is going to be selected as the high-level waste repository, and there's considerable technical controversy still over suitability."	The SBSNF EIS does not assume that Yucca Mountain will be selected as the high-level waste repository. It only assumes that, at some time in the future, a geologic waste repository will be licensed and operated by DOE and will receive spent nuclear fuel and high-level radioactive waste.
801-6	"If Yucca Mountain is chosen, the final waste acceptance criteria have not yet been established and there's a programmatic risk, as the DEIS states, that the final waste forms won't meet whatever criteria are chosen."	See response to comment 801-3.
801-7	"The argument in the EIS that potential waste forms should be developed in parallel with the repository is inconsistent with the fact that processing will start in the year 2000. This is five years before the estimated time for receiving a construction permit from the NRC [U.S. Nuclear Regulatory Commission], which will be a necessary step in developing the final waste form. You're actually proposing to process this spent fuel, not develop potential waste forms, as it states in the purpose and need for action. And these are not parallel processes; these are sequential processes, with one coming very much before the other and in my opinion, the wrong order."	The siting and development of a repository, the finalization of the waste acceptance criteria, and the treatment and management of sodium-bonded spent nuclear fuel are not necessarily sequential actions, but are interdependent parts of a larger action outlined in the Record of Decision for the Programmatic Spent Nuclear Fuel EIS (60 FR 28680). The relationship between this EIS and these interdependent actions is discussed and addressed, where appropriate, in the EIS. As stated in Section 1.2 of the EIS, DOE considers it prudent to evaluate the alternative technologies now while it is performing site characterization activities for the potential repository at Yucca Mountain. Also, to ensure the State of Idaho Settlement Agreement and Consent Order is met, and to facilitate disposal, DOE needs to reduce the uncertainties associated with qualifying sodium-bonded spent nuclear fuel for disposal. Appropriate treatment and management of the sodium-bonded spent nuclear fuel would significantly reduce complications related to disposal qualifications. The borosilicate glass waste form resulting from PUREX processing has been extensively tested and analyzed under conditions relevant to a geologic repository. DOE expects that other forms (e.g., ceramic, metallic, and high-integrity cans that do not contain metallic sodium) would be suitable for repository disposal. The development of waste forms in parallel with the development of the repository is one of many considerations discussed under the purpose and need section of the EIS (see Section 1.2). The primary consideration is the removal or conversion of metallic sodium to a nonreactive form. See response to comment 801-3.

Comments from the Arlington, Virginia, Public Hearing August 31, 1999		
No.	*Comments*	*DOE Responses*
801-8	"There also are no immediate time constraints posed by the State of Idaho settlements. As I said earlier, you know, spent fuel doesn't have to be removed until 2035. Even if you take a certain number of years to develop alternative processing, if so desired, and a certain number of years to process those, 2035 is a long ways off still."	See response to comment 801-3.
801-9	"I think it needs to be clear in the EIS that, of 60 metric tons of this spent fuel, as you stated earlier, 57 metric tons can have the sodium removed without any of these proposed processes. And also that these 57 metric tons also don't contain any HEU, which is another issue stated in the EIS as a purpose and need for action."	The EIS, under Alternative 2 (Section 2.5.3), analyzes the environmental impacts of removing sodium from 57 metric tons of blanket sodium-bonded spent nuclear fuel and the subsequent packaging of this fuel in high-integrity cans without any additional treatment and/or stabilization of the spent nuclear fuel. The environmental consequences of this action are presented in Section 4.4. As described in Appendix D, Section D.3.2.2, the uranium in the 57 metric tons of blanket fuel is depleted uranium and not highly-enriched uranium. Section 2.2 of the EIS was revised to be consistent with the information presented in Appendix D. If the finalized waste acceptance criteria for the repository requires the removal of sodium from the spent nuclear fuel, this requirement would apply to all 60 metric tons of sodium-bonded spent nuclear fuel addressed in this EIS. As described in Sections 2.2 and 2.3.9 of the EIS (formerly Section 2.4.9 of the draft EIS issued in July 1999), different treatment methods are required for the removal of sodium from driver fuel (3 metric tons) and blanket fuel (57 metric tons).

Comments from the Arlington, Virginia, Public Hearing *August 31, 1999*		
No.	*Comments*	*DOE Responses*
801-10	"So really what we're talking about is three metric tons in terms of the sodium removal, and possibly another 57 metric tons in terms of the uranium issues. But that needs to be clear and needs to be stated under what criteria those would be an issue in terms of the repository."	As discussed in Section 1.2 of the EIS, the need for the proposed action is to ensure that the requirements of the State of Idaho Settlement Agreement and Consent Order are met and to facilitate disposal of the sodium-bonded spent nuclear fuel in a geologic repository. The need for this facilitation is the reduction of the programmatic risk associated with the presence of metallic sodium, the presence of metallic uranium or highly enriched uranium in the spent nuclear fuel, and the ongoing development of high-level radioactive waste acceptance criteria for repository disposal. The goal of each of the reasonable alternatives evaluated in the EIS is to reduce the programmatic risk in different ways. The commentor's assertion that the treatment of driver spent nuclear fuel is about sodium removal and the treatment of blanket spent nuclear fuel, beyond sodium removal, is about other issues discussed in the purpose and need section of the EIS is correct. For example, Alternative 2 in the EIS addresses only sodium removal. The other alternatives go beyond sodium removal. It should be noted that PUREX processing at SRS was included as a reasonable alternative in response to the National Research Council recommendation that only PUREX processing would provide a viable alternative to the electrometallurgical treatment technology. DOE believes that the EIS is clear on the issues related to the waste acceptance criteria for repository disposal.
801-11	"So not only have, you know, you not necessarily made the case, at least in our opinion, as to why you need to do this now and what the purpose is of this process...,"	See response to comment 801-3.

Comments from the Arlington, Virginia, Public Hearing August 31, 1999		
No.	*Comments*	*DOE Responses*
801-12	"...there are a number of proliferation risks which have not been brought up yet in this meeting, which I'm a little disappointed. So let me discuss just really briefly this nonproliferation review that was put out. It does note a few of the important proliferation risks posed by EMT. And you know, it can produce weapons-usable HEU. It is a subset of a larger process which can separate plutonium and, therefore, it parallels with traditional reprocessing techniques. It does involve both processing, which makes international safeguards harder to implement, and safeguards have not been demonstrated on this technology. I'm not going to go into very much detail since there is nobody here from the Non-Proliferation Office. Let me state though, that review does underplay a lot of the risks of EMT in particular. And I focus on EMT simply because it is such a major portion of this EIS, despite the fact of the name change and the addition of other proposed actions. This started off as an EMT EIS. EMT is a major part of why these are alternatives." You know, the fact that DOE concludes in this review that EMT fully maintains consistency with U.S. nonproliferation policy is very puzzling to me considering its potential implications, both in the U.S. And globally. ...So as I say, I'm not going to go through a lot of these other nonproliferation comments, since they don't seem relevant here, but let me just note that, in terms of EMT, something that needs to be really taken into consideration is the fact that it is a process which is a subset of pyroprocessing, which could have the cadmium cathode and cathode processor put back in. You'd then end up with a substance—once you've removed that cadmium cathode and processed it—which is up to 70 percent plutonium. If a proliferator decided to then take that plutonium product—70 percent plutonium, about 30 percent uranium, less than one percent fission products, according to the OTA study from '94, and I imagine those numbers haven't changed all that much—an aqueous process to then separate out the plutonium from that would be a much different aqueous process than international safeguards are used to dealing with..." Much smaller scale of materials to be processed [*sic*]. You don't have the fission products to worry about. Yes, you have a bit of a higher radiation dose than separated plutonium, but a poor Asian country is not going to worry about that. So I think you've got to be clear as to what the implications of this are in that nonproliferation review. It kind of was a bit of a whitewash."	The assessment of nonproliferation impacts is not part of the scope of the EIS. However, none of the alternatives analyzed in this EIS, except PUREX processing at SRS, would generate weapons-usable fissile materials. Although highly enriched uranium would be an interim product, it would be down-blended to low-enriched uranium during electrometallurgical treatment. There are several features of the electrometallurgical treatment process that make it adaptable to international safeguards. The process cell, made inaccessible to humans by high radiation, inert atmosphere, and thick concrete walls, has a minimal number of penetrations through which materials can be moved in and out. These openings are secured and can be readily monitored for material transfers. There are no liquid waste streams through which materials can be piped out of the facility. All by-products and waste from the process would be in solid form, and so would be accountable by unit inventory. Finally, all by-products and waste moving out of the facility could be subjected to nondestructive examination if additional assurances were required under international safeguards agreements. As conceived for the canceled Integral Fast Reactor project, the liquid cadmium cathode would have produced a metal alloy product containing up to 70 percent plutonium which could only have been obtained after subsequent processing in a high-temperature vacuum furnace. The balance of materials would be those elements most difficult to separate from plutonium by any chemical means, such as uranium, americium, neptunium, curium, and the rare earth fission products. The plutonium metal-alloy product would have high fission product and transuranic content, a high heat source, a high neutron radiation source, and a high gamma radiation source, any one of which would make design of a weapon extremely difficult. Neutron and gamma radiation would be three to four orders of magnitude higher than weapons-grade or reactor-grade material. These levels of radiation are lethal and would prohibit any handling of the material or weapon by other than remote means. Development of the cathode progressed only to the point of technical feasibility. No prototype or working model was ever commissioned for the Fuel Conditioning Facility. Under the electrometallurgical treatment process, plutonium would stay mixed with the fission products and electrolyte salt. Plutonium and fission products would then be immobilized in the ceramic waste form. The ceramic waste form is more resistant to plutonium recovery than the metallic forms that result from other alternatives that use the melt and dilute process and high-integrity cans.

Comments from the Arlington, Virginia, Public Hearing August 31, 1999		
No.	*Comments*	*DOE Responses*
Edwin Lyman		
802-1	"I would like to commend the people in charge of this process for responding, I think, really in a surprising way to some of the comments that Nuclear Control Institute and others made during the scoping process. Restructuring the shape of the EIS so that at least the title didn't reflect—the emphasis on Electrometallurgical treatment was a pleasant surprise, as well as the acknowledgment more explicitly that the characteristics of the blanket and the driver were different; that the blanket which formed the bulk of the fuel could have the sodium removed much more simply than the driver fuel."	The comment is noted. DOE revised the scope of the EIS based on comments provided during the public scoping period.
802-2	"And even the acknowledgment that the option that involves mechanical decladding and sodium removal, the blanket, seems to be cheaper according to the Cost Study, which is another pleasant surprise, but something we might have anticipated."	Actual costs for treating and managing the sodium-bonded spent nuclear fuel are not part of the scope of the EIS. However, according to the August 1999 Cost Study, the least expensive alternative to No Action is Alternative 2, which includes blanket spent nuclear fuel sodium removal, but does not include mechanical decladding. Information such as costs, schedules, environmental consequences, and technical risk will factor into the Record of Decision for the treatment and management of sodium-bonded spent nuclear fuel.

Comments from the Arlington, Virginia, Public Hearing August 31, 1999		
No.	*Comments*	*DOE Responses*
802-3	"That said, I don't think that the draft EIS in its present form really addresses the key issue which has come up before and I'd like to reiterate it; the fact that if you're only looking now at three tons of fuel there has been no demonstration other than hand-waving referring to draft waste acceptance criteria, referring to RCRA; why this fuel cannot be directly disposed of in any repository being that it's such a small fraction of the overall inventory of radionuclides in the repository. ...I'm not advocating that corners be cut on safety, but I'd say we haven't seen a demonstration yet of why this small amount of sodium-bonded fuel would actually contribute in a significant way to the overall environmental consequences of the repository."	The uncertainties associated with qualifying sodium-bonded spent nuclear fuel for repository disposal are based on the existing regulatory environment. As discussed in Section 4.12.1 of the EIS, one of the key Nuclear Regulatory Commission requirements for acceptance of spent nuclear fuel or high-level radioactive waste is that it cannot contain or generate materials that are explosive, pyrophoric, or chemically reactive (in a repository environment) in a form or amount that could compromise the repository's ability to perform its waste isolation function or to satisfy its performance objective (10 CFR 60.135(b)(1)). In addition, in accordance with the April 1999 version of the DOE Office of Civilian Radioactive Waste Management's Waste Acceptance Systems Requirements Document, only spent nuclear fuel and high-level radioactive waste that is not subject to regulation under RCRA, Subtitle C, and meets all other acceptance criteria (e.g., packaging, uranium content), will be accepted for disposal. Although this determination for sodium-bonded spent nuclear fuel has not been made, it is a possible outcome. Based on the current regulatory environment, it is highly probable that sodium-bonded spent nuclear fuel will not be qualified for repository disposal without removal or conversion of the metallic sodium to a nonreactive form.

Comments from the Arlington, Virginia, Public Hearing August 31, 1999		
No.	*Comments*	*DOE Responses*
802-4	"And I'd just like to point out that DOE seems to embrace certain risk constraints when it sees fit and try to amend or seek waivers for others, and just comparing Yucca Mountain and WIPP makes it pretty clear.... I just read that DOE is now proposing shipping sand slag and crucibles from Rocky Flats directly to WIPP despite the fact that it contains a variety of reactive metals in it and it's going to seek a waiver for any safety issues associated with that. ...So it seems that these rules can be bent when it's feasible."	While the commentor's opinion about DOE embracing risk constraints when appropriate or seeking waivers for safety issues involving waste disposal is noted, the comment is beyond the scope of this EIS. The commentor also makes reference to the shipment of sand slag and crucibles from Rocky Flats directly to the Waste Isolation Pilot Project, which is also outside the scope of the EIS. However, in response to the commentor's statement, DOE would like to note the following activities regarding the shipment of sand, slag, and crucible residues to the Waste Isolation Pilot Project that were completed in 1999: (1) In July, after conducting a sampling analysis of the sand, slag, and crucible residues, DOE concluded there would be no pyrophoric hazards with this material. The analysis showed that these residues are sufficiently nonreactive to be shipped to the Waste Isolation Pilot Project. (2) DOE obtained the U.S. Nuclear Regulatory Commission approval in June 1999 for a change to shipping codes for the movement of material to the Waste Isolation Pilot Project. This revision allows DOE to ship residues with a passivated calcium constituent greater than that present in the sand, slag, and crucible residues. Basically, it has been determined that the sand, slag, and crucible residues are not hazardous waste and, therefore, are not subject to RCRA regulations. DOE has concluded, with the U.S. Nuclear Regulatory Commission approval, that disposal of these types of residues at the Waste Isolation Pilot Project will not adversely affect public health and safety.
802-5	"I'd like to see actual laboratory leach studies on samples of this fuel to see how this sodium, the residual sodium, and the driver fuel actually is [*sic*] released in the chemical form if you actually have the kinds of violent and potentially explosive reactions that are postulated. There's nothing like that in this document."	As discussed in Section E.4.6, the EBR-II fuel at INTEC's Basins 666 and 66 are stored inside sealed stainless steel cans that prevent the contact of basin water with the fuel cladding. During the average 17 years of storage in Basin 666, 10 of the 2148 cans were confirmed to have water in-leakage. With water inside these cans, a fuel-water reaction had produced hydrogen gas, which created bubbles that allowed detection of the water. These observations are consistent with the fact that sodium and metallic uranium react with water to produce hydrogen and this is the reason that all the sodium-bonded spent nuclear fuel is stored in dry storage or sealed containers that prevent the exposure of the fuel cladding to water. In a storage condition in a geologic repository, fuel cladding could disintegrate over time, leading to the collection of a large amount of sodium within the confines of the storage can. If this fuel can were to fail, a large amount of sodium would be available to react with any water in the repository. This could result in a violent reaction. DOE considers this condition to be unacceptable. The EIS, under the No Action alternative, analyzed a direct disposal option that was conditional on the acceptability of untreated sodium-bonded spent nuclear fuel in a repository. However, the feasibility and acceptability of such action remains to be determined.

Comments from the Arlington, Virginia, Public Hearing August 31, 1999		
No.	*Comments*	*DOE Responses*
802-6	"And as a matter of fact, in its evaluation of the No Action Alternative you refer to the fact that you're going to look at the question of the repository—of direct disposal of unprocessed driver fuel—and yet there's no mention of it other than we're going to do it. There's no discussion. And then that really has to be a key part. Because now we're talking about a very small amount of material [in comparison to overall inventory of the repository]."	The environmental impacts of the direct disposal of driver and blanket sodium-bonded spent nuclear fuel are discussed in Section 4.2 of the EIS. This is the option in which the sodium-bonded spent nuclear fuel would be disposed of in a geologic repository without sodium removal. Before the waste acceptance criteria are finalized, it is difficult to know whether this option is viable. It is possible, depending on how the final criteria are expressed, to demonstrate that, although metallic sodium is reactive and ignitable, its presence does not give the same characteristics to the sodium-bonded spent nuclear fuel and, therefore, untreated driver fuel could meet the criteria. As discussed in Section 2.5 of the EIS, DOE could decide on a hybrid alternative that includes no action for the driver fuel in the Record of Decision.
802-7	"Moving on, so in that regard, you also don't evaluate the option of mechanical sodium removal for the blanket fuel and direct disposal of the driver fuel. That is not one of the options that's considered and I think it should be. Right now—in other words, the No Action— combining the No Action Alternative and the Alternative Two should be another one that's considered."	As discussed in Section 2.5 of the EIS, DOE considered the separate treatment of the driver and blanket spent nuclear fuel in identifying a preferred alternative. DOE will consider this separate treatment in the Record of Decision. The environmental impact analyses in the EIS allow DOE to consider all combinations of technologies, options, and fuel types, including combinations not included among the specific combinations explicitly analyzed in the EIS. As the commentor suggests, "no action" could be considered for the driver spent nuclear fuel, and "high-integrity can packaging" for the blanket spent nuclear fuel.
802-8	"I'd just like to point out a few other inconsistencies, or just one. For instance, the uranium which is recovered from the Electrometallurgical treatment of the fuel. This is not being credited with a— it does not have a value according to the Cost Study, which is reasonable because DOE is not going to be selling any of its uranium for 10 years to support the market price in the context of the U.S.-Russian Agreement." However, you then do not consider it part of the waste stream and, since Anna Aurillo isn't here and she likes to reiterate this issue, it should be, especially if it's not a commodity that has a value. If you can't sell it, then it's a waste, and so the volume associated with that should certainly be added to the table."	The uranium recovered from the electrometallurgical treatment process contains radioactive isotopes that render it unusable as surplus uranium without further processing to remove these impurities. DOE has not yet determined the final disposition of this uranium. For the purpose of the EIS, it is assumed that metal uranium ingots from the electrometallurgical treatment process would be stored in the Materials Building within the Zero Power Physics Reactor at ANL-W. The uranium recovered from the electrometallurgical treatment process has not been treated as a waste because of its potential value if it is further processed. This uranium will be categorized when DOE determines if it will be further processed.

A.2.6 Written Comments and DOE Responses

Comments presented in this section were submitted to DOE via the U.S. mail, e-mail, toll-free number, toll-free fax line, or in person at the public hearings. All comments received during the comment period, which began on July 31, 1999, and ended on September 28, 1999, as well as submittals received after September 28, are reproduced in this section. This section provides a side-by-side display of the written comments received (full-text reproductions) and DOE's responses. Individual comments are numbered in the margins of the comment letters, and DOE responses to each of the numbered comments are provided on the right side of each page.

Commentor No. 1: Ellen Glaccum

Draft EIS Comment Form

Treatment and Management of Sodium-Bonded Spent Nuclear Fuel

I find it mystifying that you have gone to the time & expense to draw a preferred alternative that is "no preferred alternative." I hereby request that you send me copies of the "independent" cost study & the non-proliferation report. I do commend you from pulling back from your former cheer-leader role for Argonne's electrometallurgical scheme. I question the timing of this project. Why not wait to see what the acceptance criteria at the mythical permanent repository will actually be, rather than guessing at this point in time? I am very interested in the fact that air emissions could be 7,162 curies of tritium and 32,250 curies of krypton-85 every year — will there be other radioactive air emissions? What & how much? What will the total air emissions at INEL be with this, the planned incinerator, and all other current & planned ventures? How will the SNF not currently located at INEL be transported to INEL? Will the public be informed abt transportation schedules? I demand the final EIS acknowledge that the "background radiation" has doubled since the Trinity testing.

1-1
1-2
1-3
1-4
1-5
1-6
1-7
1-8

There are several ways to provide comments on the Draft EIS for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel and these include: nothing but results

- attending public hearings and giving your comments directly to DOE representatives
- returning this comment form to the registration desk at a public hearing or to the address listed below
- calling toll-free and leaving your comments: 1-877-450-6904
- faxing your comments toll-free to: 1-877-621-8288
- commenting via e-mail: emteis@hq.doe.gov

Name (optional): Ellen Glaccum

Organization: citizen / taxpayer

Home/Organization Address (circle one): Box 1173

City: Ketchum State: ID Zip Code: 83340

Telephone (optional): _____

COMMENTS MUST BE POSTMARKED BY SEPTEMBER 13, 1999

For more information contact: Susan Leeka, NE-60
U.S. Department of Energy • 19901 Germantown Road • Germantown, MD 20874
Toll-free Telephone: 1-877-450-6904 • Toll-free Fax: 1-877-621-8288
E-mail: emteis@hq.doe.gov • Website: http://www.ne.doe.gov/home/eis.html

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06/10/99

Response to Commentor No. 1:

1-1: Council on Environmental Quality regulations (40 CFR 1502.14[e]) do not require a preferred alternative to be included in a draft EIS if one has not been identified at the time of publication. However, the regulations do require that a preferred alternative be identified in a final EIS. DOE initially identified electrometallurgical treatment at ANL-W as the Preferred Alternative in its Notice of Intent (64 FR 8553). However, in response to public comments received during the scoping period, a preferred alternative was not identified in the draft EIS. This was done so that the EIS would better reflect a broader range of potential treatment alternatives. Section 2.8 of this EIS identifies Alternative 1, electrometallurgical treatment, as the Preferred Alternative for the proposed action.

1-2: Copies of the Cost Study and Nonproliferation Impacts Assessment were sent to the commentor. These reports were mailed to all interested parties on August 12, 1999, during the comment period and were also made available at the public hearings on the draft EIS. Although these reports are not critical to the evaluation of the analysis presented in the EIS, they will be considered during the decision-making process in the preparation of the Record of Decision.

1-3: DOE initially identified electrometallurgical treatment at ANL-W as the proposed action in its Notice of Intent (64 FR 8553). However, in response to public comments received during the scoping period, a preferred alternative was not identified in the draft EIS. This was done so that the EIS would better reflect a broader range of potential treatment alternatives.

1-4: Although the waste acceptance criteria have not been finalized, there is substantial guidance provided by DOE's Office of Civilian Waste Management in their "Waste Acceptance System Requirements Document," Revision 3, April 1999, which is referenced in the EIS. Based on this guidance (see Section 4.12.1 of the EIS), it is highly probable that sodium-bonded spent nuclear fuel would not be acceptable in the repository without some stabilization and/or removal of the metallic sodium. Having successfully completed the Electrometallurgical Treatment Research and Demonstration Project (see Section 1.6.3) and in planning the closure of its PUREX processing capabilities, DOE needs to decide whether these processes are suitable for treating the remaining sodium-bonded spent nuclear fuel, or whether there is sufficient reason to delay a decision and wait for the development of other treatment technologies. Delaying the EIS could result in a loss of capability and of experienced, knowledgeable technical staff, should DOE decide at a later date to use the electrometallurgical

Commentor No. 1: Ellen Glaccum

Response to Commentor No. 1 (Cont'd):

process to treat sodium-bonded spent nucler fuel. Section 1.2 of the EIS has been revised for clarification.

1-5: The maximum annual radiological gaseous (air) emissions would occur during simultaneous melt and dilute processing of the EBR-II driver and blanket spent nuclear fuel under Alternative 6. This simultaneous operation would occur over two years. The estimated total curies released during treatment of the sodium-bonded spent nuclear fuel at ANL-W under Alternative 6 would be about 4,300 curies of elemental tritium and about 67,000 curies of krypton-85. As indicated in the EIS, the radiological dose impacts from these releases to the general public residing within 80 kilometers (50 miles) of the facility would be well below regulatory limits. These two radionuclides (tritium and krypton-85) would account for greater than 99.9 percent of estimated dose to the population. Appendix E of the EIS lists all potential radionuclides that could be released by the proposed action. As indicated in this appendix, other airborne releases would be orders of magnitude smaller than these two nuclides. After two years, the krypton and tritium releases would be 520 and 70 curies per year, respectively. Overall, the radiological impacts associated with these releases would result in individual maximum doses much smaller than the 10 millirem per year limit set by the EPA for radioactive air emissions under 40 CFR 61.

1-6: As explained in Section 3.2.3.1 of the EIS, total releases of tritium and krypton-85 at INEEL from all operations during 1997 (the most currently available data) resulted in approximately 430 and 3,580 curies, respectively. The planned incinerator at INEEL, which was evaluated under the Advanced Mixed Waste Project Final EIS (DOE/EIS-0290), is expected to produce about 27 curies of tritium and a very small amount of krypton-85 per year. Releases during other, proposed and planned activities for the future are documented in various EISs that are listed in Section 1.6 of this EIS. Maximum impacts from air emissions associated treatment of the sodium-bonded spent nuclear fuel and those of future activities at INEEL are summarized in Section 4.11.1.4 of the EIS. The results clearly indicate that the cumulative impacts (collective doses to the maximally exposed offsite individual and the general public over the duration of the operation) from the expected releases would be well below the regulatory limit.

1-7: Sodium-bonded spent nuclear fuel not currently located at INEEL will be transported to INEEL in accordance with the amended Record of Decision for the DOE Programmatic Spent Nuclear Fuel EIS (61 FR 9441). All information regarding the transport of this spent nuclear fuel will be

Commentor No. 1: Ellen Glaccum

Response to Commentor No. 1 (Cont'd):

disseminated in accordance with the programmatic EIS and is not considered part of the scope of this SBSNF EIS. This is discussed in Section 4.9 and Appendix G of this EIS. DOE will inform the state and Tribal governments about transportation schedules regarding the spent nuclear fuel addressed in this EIS.

1-8: As indicated in Appendix E, Section E.2.1, an average American would receive about 300 millirem per year from cosmic, terrestrial (Earth's rock formations), and natural (radon gas) radiation sources. The background radiation dose from atmospheric bomb tests (including the Trinity testing) is a fraction of 1 millirem per year.

Commentor No. 2: Richard Albrecht

Treatment and Management of Sodium-Bonded Spent Nuclear Fuel

Draft EIS Comment Form

7/27/99

I AM IN FAVOR OF THE OPERATION OF THE TREATMENT AND MANAGEMENT OF SODIUM-BONDED SPENT NUCLEAR FUEL FACILITY PROPOSED FOR THE INEEL IN IDAHO.

IT'S TIME FOR US TO LIFT OUR HEADS OUT OF THE FEAR OF MISUNDERSTANDING REGARDING NUCLEAR ENERGY OPERATION AND CONTROL. WE MUST LOOK AHEAD INTO THE NEXT CENTURY WITH INTELLIGENT THOUGHTS AND CONSIDERATION. EMOTIONAL AND IGNORANT APPEALS TO INSPIRE TOTAL NEGATIVE RESPONSE ARE NO LONGER ACCEPTABLE AS REASONABLE ARGUMENTS WHEN ALL OF OUR FUTURE IS AT STAKE.

PLEASE PLACE IMPORTANCE ON FACTUAL REASON AND LOGIC RATHER THAN A FABRICATED EMOTIONAL PREJUDICE BASED ON FEAR AND IGNORANCE.

2-1

There are several ways to provide comments on the Draft EIS for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel and these include:

- attending public hearings and giving your comments directly to DOE representatives
- returning this comment form to the registration desk at a public hearing or to the address listed below
- calling toll-free and leaving your comments: 1-877-450-6904
- faxing your comments toll-free to: 1-877-621-8288
- commenting via e-mail: emteis@hq.doe.gov

Name (optional): RICHARD ALBRECHT

Organization: _____

Home/Organization Address (circle one): P.O. Box 497

City: WILSON, State: WY Zip Code: 83014

Telephone (optional): _____

COMMENTS MUST BE POSTMARKED BY SEPTEMBER 13, 1999

For more information contact: Susan Lesica, NE-40
U.S. Department of Energy • 19901 Germantown Road • Germantown, MD 20874
Toll-free Telephone: 1-877-450-6904 • Toll-free Fax: 1-877-621-8288
E-mail: emteis@hq.doe.gov • Website: http://www.ne.doe.gov/home/eis.html

06/10/99

Response to Commentor No. 2:

2-1: The commentor's support for the treatment and management of sodium-bonded spent nuclear fuel at INEEL is noted.

Commentor No. 3: Peter J. Dirkmaat

Treatment and Management of Sodium-Bonded Spent Nuclear Fuel

Draft EIS Comment Form

(1) THE SUMMARY STATES ON P. S-35 THAT HAZARDOUS CHEMICAL EXPOSURES WOULD BE MINIMAL UNDER NORMAL OR ACCIDENT CONDITIONS. SINCE THE MOLTEN SALT BATH CONTAINS CADMIUM AND MAYBE OTHER HAZARDOUS CONSTITUENTS, IT IS NOT CLEAR WHAT PRECAUTIONS ARE TAKEN TO PROTECT WORKERS AND THE PUBLIC DURING PREPARATION, PROCESSING, AND DISPOSAL OF THE MOLTEN SALT. ADDITIONAL DISSUSSION IS WARRENTED. 3-1

(2) THE PREVIOUS NA-BONDED CLEANING EXPERIENCE OF 17 MTHM OF EBR-II DRIVER FUEL AT ROCKETDYNE SHOULD BE ACKNOWLEDGED WITH DISCUSSION OF WHY SUCH PROCESS IS NOT A DESIRABLE OPTION. 3-2

There are several ways to provide comments on the Draft EIS for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel and these include:

- attending public hearings and giving your comments directly to DOE representatives
- returning this comment form to the registration desk at a public hearing or to the address listed below
- calling toll-free and leaving your comments: 1-877-450-6904
- faxing your comments toll-free to: 1-877-621-8288
- commenting via e-mail: emteis@hq.doe.gov

Name (optional): PETER J. DIRKMAAT

Organization: DOE-ID

Home/Organization Address (circle one): 785 DOE PLACE
IDAHO FALLS, ID 83401

City: SHELLEY State: ID Zip Code: 83274

Telephone (optional): (208) 526-1439

COMMENTS MUST BE POSTMARKED BY SEPTEMBER 13, 1999

For more information contact: Susan Leckie, NE-40
U.S. Department of Energy • 19901 Germantown Road • Germantown, MD 20874
Toll-free Telephone: 1-877-450-6904 • Toll-free Fax: 1-877-621-8288
E-mail: emteis@hq.doe.gov • Website: http://www.ne.doe.gov/home/eis.html

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06/10/99

Response to Commentor No. 3:

3-1: In one of the electrorefiner designs for the electrometallurgical treatment of sodium-bonded driver spent nuclear fuel contains a layer of cadmium to allow recovery of the uranium that falls off the cathode during treatment. This electrorefiner design provides a cadmium vapor trap that collects, condenses, and returns any cadmium vapor generated during operation. In addition, ANL-W has incorporated cadmium worker safety in its operations through administrative procedures and worker training. Therefore, the workers are considered to be protected from cadmium hazards. The only abnormal condition that could lead to accidental releases of cadmium in the hot cell and the environment is hypothesized in the EIS to occur during a beyond-design-basis earthquake with an estimated frequency of 0.00001 per year. Given such an earthquake, the EIS estimates the consequences of a cadmium release to the noninvolved worker would be orders of magnitude lower than the Emergency Response Planning Guideline-1 (ERPG-1) value, so it would have a minimal impact.

3-2: The sodium cleaning process used at Rocketdyne and the reasons why this process was not explicitly evaluated in the EIS are described in revised Section 2.3.9 and Section C.2 of Appendix C of the EIS.

Commentor No. 4: Susan Pengilly Neitzel

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Boise, Idaho 83702-5642
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Fax: (208) 334-2774

Archaeological Survey
210 Main Street
Boise, Idaho 83702-7264
Office: (208) 334-3847
Fax: (208) 334-2775

Historical Museum and Education Programs
610 North John Davis Drive
Boise, Idaho 83702-7695
Office: (208) 334-2120
Fax: (208) 334-4059

Historic Preservation Office
210 Main Street
Boise, Idaho 83702-7264
Office: (208) 334-3861
Fax: (208) 334-2775

Historic Sites Office
2445 Old Penitentiary Road
Boise, Idaho 83712-8254
Office: (208) 334-2844
Fax: (208) 334-3225

Library/Historical Collection
450 North Fourth Street
Boise, Idaho 83702-6027
Office: (208) 334-3356
Fax: (208) 334-3198

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450 North Fourth Street
Boise, Idaho 83702-6027
Office: (208) 334-3357
Fax: (208) 334-3198

Oral History
450 North Fourth Street
Boise, Idaho 83702-6027
Office: (208) 334-3863
Fax: (208) 334-3198

Memberships and Outreach and Development
1109 Main Street, Suite 230
Boise, Idaho 83702-5642
Office: (208) 334-3986
Fax: (208) 334-2774

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Boise, Idaho 83702-6027
Office: (208) 334-3428
Fax: (208) 334-3198

State Archives/Manuscripts
2205 Old Penitentiary Road
Boise, Idaho 83712-8250
Office: (208) 334-2620
Fax: (208) 334-2626

July 29, 1999

Ms. Susan Lesica
EIS Document Manager
Office of Nuclear Energy
Science and Technology (NE-40)
19901 Germantown Road
Germantown, Maryland 20874-1290

RE: Treatment and Management of Sodium-Bonded Spent Nuclear Fuel, Argonne National Laboratory-West, Idaho National Engineering and Environmental Laboratory, Idaho

Dear Ms. Lesica:

Thank you for sending the draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel. The project may take place at Argonne National Laboratory-West.

We are concerned about any actions associated with the project that may affect potentially significant buildings and structures located at ANL-West. If interior or exterior alterations to the existing facilities are planned, we will need to receive a report containing the historic context of ANL-West and documentation of the individual facilities. We will also be requesting a final draft of the historic context for DOE-Id facilities at the INEEL for reference in evaluating properties located at ANL-West.

4-1

To prepare the report, we recommend that you contract with a cultural resource professional who meets the Secretary of the Interior's professional qualifications for historian or architectural historian. Robert Stark and the cultural resource staff at Lockheed-Martin will be able to assist you in identifying a professional contractor.

We appreciate your cooperation. If you have any questions, feel free to contact me at 208-334-3827.

Sincerely,

Susan Pengilly Neitzel

Susan Pengilly Neitzel
Deputy SHPO and
Compliance Coordinator

cc: Robert Stark, DOE-Id

The Idaho State Historical Society is an Equal Opportunity Employer.

Response to Commentor No. 4:

4-1: DOE has examined all reasonable alternatives that involve facilities at ANL-W, and none have been found that would have an adverse affect on the interior or exterior of any facility at the site. The alternatives vary primarily by the type of equipment that would be installed inside the Fuel Conditioning Facility, the Hot Fuel Examination Facility, and other facilities at ANL-W. There are, therefore, no alterations planned that would change the historic value of these buildings. Thus, an ANL-W historic context report is not required for the proposed action described in the EIS.

Commentor No. 5: Pat Clark

SB SNF Toll Free Line

8/5/99

Pat Clark
Snake River Alliance
hm: 208-344-3932
Snake River Alliance: 208-344-9161

I'm a concerned, well I'd like some information on the hearings that are coming up on the draft EIS of the sodium treatment. What I'd like to know is I'd like to ask if we can have a 60 day extension and I'd like to know who to direct this question to. **5-1**

Response to Commentor No. 5:

5-1: In an effort to ensure that all interested parties had time to comment on the draft EIS, the deadline for transmittal of comments was extended from September 13 to September 28, 1999 (64 FR 49169).

Commentor No. 6: Charles Bailey

SB SNF Toll Free Line

8/6/99

Charles Bailey
803-725-4435

I'd like to make a public comment about having the stabilization material process through the SRS F-Canyon with the melt and dilute process in the 105-L area due to the fact that we have the infrastructure in place. We have community and public support. We have Congressional and political support and we can do it cost effective and more importantly we can do it safely, more so than any other site can possibly imagine. This is my feelings on the subject and I just wanted to call and let you people know. Thank you and have a good day. **6-1**

Response to Commentor No. 6:

6-1: The commentor's support for the treatment and management of sodium-bonded spent nuclear fuel at SRS is noted.

Commentor No. 7: Jean Boyles

SB SNF Toll Free Line

8/6/99

Jean Boyles
208-343-0919

I'd like to request a 60 day extension for the comment period. Yeah, we need more information needs to be gathered. 7-1

Response to Commentor No. 7:

7-1: In an effort to ensure that all interested parties had time to comment on the draft EIS, the deadline for transmittal of comments was extended from September 13 to September 28, 1999 (64 FR 49169). With respect to the need for more information, DOE obtained and analyzed the relevant information and made that information available to the public. Background materials were placed in public reading rooms and were made available to the public through a series of hearings held August 17, 1999, in North Augusta, South Carolina; August 24, 1999, in Boise, Idaho; August 26, 1999, in Idaho Falls, Idaho; and August 31, 1999, in Arlington, Virginia. Materials placed in the reading rooms included the electrometallurgical demonstration project environmental assessment, the Finding of No Significant Impact for the environmental assessment, National Research Council interim status reports on the electrometallurgical treatment demonstration project, the 1995 Settlement Agreement and Consent Order with the State of Idaho, the scoping meeting transcripts and comments, and the draft EIS hearing presentations and fact sheets. In addition, completion of the Cost Study and Nonproliferation Impacts Assessment was expedited so that they would be available to the public during the comment period. These reports were mailed to interested parties on August 12, 1999, and were made available to attendees at all of the public hearings on the draft EIS. Although these reports are not critical to the environmental impact analysis presented in the EIS, they will provide input to the Record of Decision. While the final National Research Council report on the demonstration project was published in April 2000, interim status reports were produced throughout the project. Data generated during the demonstration project were used in preparing the EIS, as discussed in Section 1.6.3 of the EIS.

Commentor No. 8: Lowell Jobe

SB SNF Toll Free Line

8/12/99

Lowell Jobe
Coalition 21
14469 N 55th East
Idaho Falls, ID 83401
208-524-7271
fax: 208-524-0998

I received the copies of the documents pertain to the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel DEIS. However, on page S-6 and S-7 they refer to two things that I was interested in and would like to receive either information as to whether it is available or copies of those documents and the two of them deal with the costs as referred to on page S-6 and also the nuclear nonproliferation items. Both of them were supposed to be expedited so they would available about the same time as the main DEIS. However, I believe since I'm working for Coalition 21 on evaluating the document I would like to receive these two which I think are an integral part of the entire problem. **8-1**

Response to Commentor No. 8:

8-1: Copies of the Cost Study and Nonproliferation Impacts Assessment were sent to the commentor. DOE did expedite completion of the Cost Study and the Nonproliferation Impacts Assessment. These reports were mailed to interested parties on August 12, 1999, and were made available to attendees at the public hearings on the SBSNF Draft EIS. These public hearings were held on August 17, 1999, in North Augusta, South Carolina; August 24, 1999, in Boise, Idaho; August 26, 1999, in Idaho Falls, Idaho; and August 31, 1999, in Arlington, Virginia. Although these reports are not critical to the environmental impact analysis presented in the EIS, they will provide input to the Record of Decision.

Commentor No. 9: Laird Irvin

SB SNF Toll Free Line

8/18/99

Laird Irvin
PO Box 2885
Ketchum, ID 83340

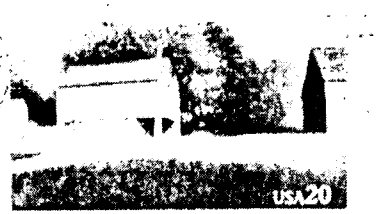
I'd like to get a 60 day extension on the comment period so we can further work on this. **9-1**

Response to Commentor No. 9:

9-1: In an effort to ensure that all interested parties had time to comment on the draft EIS, the deadline for transmittal of comments was extended from September 13 to September 28, 1999 (64 FR 49169).

Commentor No. 10: Betina Mattesen

B. Mattesen
434 Lower Notch
Bristol, VT 05443



Ms Susan Lesica
US Dept of Energy, Office of Nuclear
Fac. Management NE 40
19901 Germantown Rd
Germantown, Maryland 20874-1290

I would like to comment on the Draft EIS for the Management of Sodium Bonded Spent Nuclear Fuel. I am concerned about the wasteful cost and environmental problems of increased nuclear waste. The US has nonproliferation goals and this technology runs counter to that. The comment period must be extended - the EIS omits cost analysis of alternatives, Nat. Academy of Science review + a plan for waste disposal, among other things. A 60 day extension is requested. Thank you. Betina Mattesen

10-1
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10-6
10-3

Response to Commentor No. 10:

10-1: Chapter 4 of the EIS presents data that demonstrates that, compared to leaving the sodium-bonded spent nuclear fuel in its current form, treatment and management of the sodium-bonded spent nuclear fuel would significantly reduce the volume of high-level radioactive waste that needs to be disposed of in a geologic repository. Cost is not part of the scope of this EIS. A Cost Study was completed and distributed to interested public members during the public comment period.

10-2: The assessment of nonproliferation impacts is not a part of the EIS process; however, it should be noted that DOE's Office of Arms Control and Nonproliferation assessed the potential nonproliferation impacts that may result from each of the alternatives and technologies analyzed in this EIS. This Nonproliferation Impacts Assessment stated that, for this specific application, all alternatives except PUREX processing at SRS are fully consistent with U.S. policy with respect to reprocessing and nonproliferation.

10-3: In an effort to ensure that all interested parties had time to comment on the draft EIS, the deadline for transmittal of comments was extended from September 13 to September 28, 1999 (64 FR 49169).

10-4: The SBSNF EIS was prepared in accordance with NEPA, Council on Environmental Quality regulations on implementing NEPA (40 CFR 1500-1508), and DOE's NEPA implementation procedures (10 CFR 1021). None of these require the inclusion of a cost analysis in an EIS. As discussed in the introduction, the basic objective of this EIS is to provide the public and DOE decision-makers with a description of the reasonable alternatives for treatment and management of DOE's sodium-bonded spent nuclear fuel and their potential environmental impacts. However, DOE has issued a separate Cost Study that analyzes and compares the costs of the alternatives analyzed in the EIS. Cost will be considered during the decision-making process in preparing the Record of Decision.

10-5: The National Academy of Sciences' National Research Council prepared interim status reports on the results of the Electrometallurgical Treatment Research and Demonstration Project that have been reviewed by DOE. The National Research Council completed their evaluation of the electrometallurgical treatment demonstration project in September 1999 and published their final summary report in April 2000. The final report findings will be considered during the decision-making process leading to the Record of Decision.

Commentor No. 10: Betina Mattesen

Response to Commentor No. 10 (Cont'd):

10-6: The EIS identifies and quantifies the volume and type of waste for each alternative. A geologic repository is planned to be completed and licensed to receive spent nuclear fuel and/or high-level radioactive waste. The EIS assumes that high-level radioactive waste and/or spent nuclear fuel from each alternative of this EIS would be sent to this geologic repository. Section 4.1.2 of the EIS discusses the planned disposition of other waste generated by the proposed action.

Commentor No. 11: Susan Mathees

SB SNF Toll Free Line

8/19/99

Susan Mathees
Ketchum, ID
208-726-3471

I am requesting a 60 day extension of the comment period for the pyroprocessing draft EIS hearing schedule. I'll comment later on the rest. But please make note that it's a 60 day extension request for the comment period. Thanks very much. **11-1**

Response to Commentor No. 11:

11-1: In an effort to ensure that all interested parties had time to comment on the draft EIS, the deadline for transmittal of comments was extended from September 13 to September 28, 1999 (64 FR 49169).

Commentor No. 12: Jeep Hardinge

SB SNF Toll Free Line

8/23/99

Jeep Hardinge
Ketchum, ID 83340
208-726-4819

I would like to request an extension on the comment period for the pyroprocessing draft EIS hearing period, comment period. And feel that there will be more information available following the closing of the scheduled comment period of September 13 and would request an extension of the comment period. Thank you.

12-1

Response to Commentor No. 12:

12-1: In an effort to ensure that all interested parties had time to comment on the draft EIS, the deadline for transmittal of comments was extended from September 13 to September 28, 1999 (64 FR 49169). With respect to the need for more information, DOE obtained and analyzed the relevant information and made that information available to the public. Background materials were placed in public reading rooms and were made available to the public through a series of hearings held August 17, 1999, in North Augusta, South Carolina; August 24, 1999, in Boise, Idaho; August 26, 1999, in Idaho Falls, Idaho; and August 31, 1999, in Arlington, Virginia. Materials placed in the reading rooms included the electrometallurgical demonstration project environmental assessment, the Finding of No Significant Impact for the environmental assessment, National Research Council interim status reports on the electrometallurgical treatment demonstration project, the 1995 Settlement Agreement and Consent Order with the State of Idaho, the scoping meeting transcripts and comments, and the draft EIS hearing presentations and fact sheets. In addition, completion of the Cost Study and Nonproliferation Impacts Assessment was expedited so that they would be available to the public during the comment period. These reports were mailed to interested parties on August 12, 1999, and were made available to attendees at all of the public hearings on the draft EIS. Although these reports are not critical to the evaluation of the analysis presented in the draft EIS, they will provide input to the Record of Decision. While the final National Research Council report on the demonstration project was published in April 2000, interim status reports were produced throughout the project. Data generated during the demonstration project were used in preparing the EIS.

Commentor No. 13: Ernest S. Chaput

ECONOMIC DEVELOPMENT PARTNERSHIP
Serving Aiken & Edgefield Counties

Fred E. Humes
Director

Comments for the Record
Treatment and Management of Sodium-Bonded Spent Nuclear Fuel
Draft Environmental Impact Statement

My name is Ernest S. Chaput and I represent the Economic Development Partnership of Aiken and Edgefield Counties, South Carolina. The Savannah River Site is located immediately adjacent to Aiken, South Carolina. The Partnership routinely reviews and provides comment on proposed Department of Energy activities which may be conducted at SRS for consistency with local capabilities and community expectations.

We have three comments regarding the subject draft EIS:

- We support Department efforts to safely manage and prepare spent nuclear fuel for disposal in the National Repository. We believe that the Federal Government has the obligation to expeditiously place these and other waste materials into forms which will provide adequate long-term protection of the public health and environment. These sodium-bonded fuels are no exception. Unless the Department has an unequivocal commitment that these fuels will be accepted in the National Repository without treatment, then **we believe that treatment is mandatory**. 13-1

- We note that two of the six alternatives included in the draft EIS include the shipment of a portion of these fuels to the Savannah River Site for treatment. Alternative three treats blanket elements in the SRS canyons by the PUREX process. Alternative five treats blanket elements in the proposed SNF Melt and Dilute facility. As policy, we do not support the shipment of waste materials to SRS unless it can be clearly demonstrated that the SRS has a significant capability advantage to perform the task which cannot be reasonable established at the generating site. That is not the case for these sodium-bonded fuels. The draft EIS identifies the electrometallurgical facility which currently exists at Argonne-West, and it appears that this facility can adequately prepare the Sodium-bonded SNF for shipment to the National Repository. Because an adequate treatment capability currently exists at Argonne-West, **we object to the shipment of these wastes to the Savannah River Site.** 13-2

- If the Department determines that shipment of blanket elements to the Savannah River Site is in the national interest, then **we strongly recommend that only the PUREX treatment option (Alternative three) be considered.** Our reasons are twofold: 13-3

  The PUREX process is currently operational at SRS, and its waste form has the highest probability of acceptance at the National Repository. The vitrification of canyon high level liquid wastes in the Defense Waste Processing Facility has been

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(803) 648-3362 • FAX (803) 641-3369 • edpsc@aol.com • http://www.edpsc.org

Response to Commentor No. 13:

13-1: The commentor's support for the treatment and management of sodium-bonded spent nuclear fuel to facilitate its disposal in a repository is noted.

13-2: The commentor's objections to the shipment of spent nuclear fuel to SRS for treatment is noted. The selection of reasonable alternatives evaluated in the EIS was made in accordance with the Council on Environmental Quality Regulations (40 CFR 1500-1508) and DOE's NEPA-related regulations (10 CFR 1021) and procedures. In addition, as discussed in Section 1.3 of the EIS, the selection of reasonable alternatives is responsive to the issues raised during the public scoping period.

13-3: The commentor's preference for the PUREX process over the melt and dilute process at SRS is noted. The environmental impacts of all potential technologies are evaluated in the EIS and these will be considered, along with the assessments in the Nonproliferation Impacts Assessment and the Cost Study, during the decision-making process prior to publication of the Record of Decision. It should be noted that, although vitrified high-level radioactive waste meets current repository waste acceptance criteria, DOE expects that other waste forms would also be acceptable. DOE does not envision a situation in which sodium-bonded spent nuclear fuel would be shipped to SRS without the assurance of its ultimate disposition.

13-4: The commentor's objection to the melt and dilute process at SRS is noted. Although the products of the melt and dilute treatment process and those of the other treatment technologies have not been evaluated using existing waste acceptance criteria, it is expected that these products will be acceptable under the final waste acceptance criteria for the geologic repository when they are available. DOE does not envision a scenario in which blanket sodium-bonded spent nuclear fuel would be shipped to SRS for treatment without the assurance of its ultimate disposition.

13-5: Congress determines how funds are allocated. DOE spends monies consistent with Congressional direction. DOE is not in a position to make the difficult tradeoffs that may be required between alternative Federal programs and spending priorities. The issue of funding for the treatment and management of sodium-bonded spent nuclear fuel is beyond the scope of the SBSNF EIS.

Commentor No. 13: Ernest S. Chaput (Cont'd)

extensively reviewed and meets Repository acceptance criteria. Thus we are **assured that the wastes brought into South Carolina have a path out to final disposition.** 13-3 (Cont'd)

The Melt and Dilute process is currently in the development phase and the proposed waste form has not been extensively reviewed for acceptance in the National Repository. The Melt and Dilute program is currently underfunded and behind schedule. The inclusion of sodium-bonded blanket materials will further complicate process development and facility operation. There is no assurance that the product form from treatment of sodium-bonded fuels by the Melt and Dilute process will be subsequently shipped to the National Repository. Thus we are faced with the possibility that **sodium-bonded fuel could be shipped to South Carolina with no path out to final disposition. This is an unacceptable situation.** 13-4

It is essential that adequate budgetary resources are provided to Savannah River to meet the incremental facility operating and processing costs of treating this fuel. **We object to consideration of any Savannah River Site option without a firm DOE commitment for incremental funding.** 13-5

In summary, we support DOE efforts to prepare the subject fuels for shipment to the National Repository; however we believe that treatment should be performed at Argonne-West if at all possible. 13-1 If Savannah River capabilities are to be considered for treatment, 13-2 then only the PUREX process should be considered and only if adequate financial resources are provided. 13-3

Thank you for the opportunity to comment on this draft EIS.

Response to Commentor No. 13 :

Commentor No. 14: Don McWhorter

Treatment and Management of Sodium-Bonded Spent Nuclear Fuel

Draft EIS Comment Form

1. The presented waste volumes do not seem technically correct. The PUREX are overstated and electrometallurgical process is understated. — 14-1
2. Why doesn't the EIS consider disposition? The PUREX could dispose the uranium via the Surplus HEU Disposition as commercial fuel. The electrometallurgical process will require additional processing before disposal. — 14-2
3. Sodium is easily removed and should not be considered technically difficult — 14-3
4. The fuel can be chopped prior to the PUREX and the S/S hulls disposed as low level waste. This option should be included in Option 3 — 14-4
5. Better describe what happens to the uranium ingots produced by the electrometallurgical process. — 14-5

There are several ways to provide comments on the Draft EIS for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel and these include:

- attending public hearings and giving your comments directly to DOE representatives
- returning this comment form to the registration desk at a public hearing or to the address listed below
- calling toll-free and leaving your comments: 1-877-450-6904
- faxing your comments toll-free to: 1-877-621-8288
- commenting via e-mail: sodium.fuel.eis@hq.doe.gov

Name (optional): Don McWhorter

Organization:

Home/Organization Address (circle one): 704 Greenwood Drive

City: North Augusta State: SC Zip Code: 29841-2007

Telephone (optional): (803) 278-0077

COMMENTS MUST BE POSTMARKED BY SEPTEMBER 13, 1999

For more information contact: Shane Lantz, NE-40

Response to Commentor No. 14:

14-1: The waste volumes given in the EIS are the final solid disposal volumes. The waste volumes generated from the PUREX processing of declad and cleaned blanket spent nuclear fuel, presented in Chapter 4 of the EIS, are consistent with those presented in the SRS Spent Nuclear Fuel Management EIS for processing similar spent nuclear fuel. For example, the SRS Spent Nuclear Fuel Management EIS estimated that the PUREX processing of about 20 metric tons of heavy metal of declad and cleaned blanket spent nuclear fuel would produce a total of 170 cubic meters of liquid high-level radioactive waste. As described for Alternative 3 in this EIS, PUREX would process about 57 metric tons of heavy metal of cleaned and declad blanket spent nuclear fuel. Therefore, it was estimated that PUREX processing would generate about 510 cubic meters (667 cubic yards) of liquid high-level radioactive waste. Section 4.5.6 of the EIS describes waste generation from the PUREX processing of cleaned and declad blanket spent nuclear fuel. Estimates of the ceramic and metallic high-level radioactive waste volumes generated during electrometallurgical treatment were based on the type of fuel, zeolite, glass frit, and process characteristics, all of which are known quantities. The volume of high-level radioactive waste generated by electrometallurgical treatment that were reported in the SBSNF EIS were based on data generated from the completed demonstration project at ANL-W.

14-2: As described in Section 2.6 of the EIS, PUREX processing would not be used to treat the sodium-bonded driver spent nuclear fuel. Treatment of cleaned (sodium removed) and declad blanket spent nuclear fuel at SRS' F-Canyon (via the PUREX process) would not generate highly enriched uranium; it would produce depleted uranium. The electrometallurgical treatment process would separate the highly enriched uranium from the driver spent nuclear fuel and would downblend it to low enriched uranium. A separate NEPA action will address the disposition of uranium.

14-3: As discussed in Appendix C, Section C.1, the products of the electrometallurgical treatment are: uranium metal ingots, metallic waste forms, and ceramic waste forms. The metallic and ceramic waste forms would be considered high-level radioactive waste and would be certified for disposal in a geologic repository in accordance with repository acceptance criteria. Although the acceptance criteria are still not finalized, it is not expected that additional processing would be required for the certification of these waste forms. The uranium metal ingots, containing

Commentor No. 14: Don McWhorter

Response to Commentor No. 14 (Cont'd):

(from the treatment of blanket fuel) are not currently considered high-level radioactive waste, and are not destined for disposal in a geologic repository. Their final disposition, further use or disposal, will be determined in a future NEPA review.

14-4: As discussed in Section 2.2 of the EIS, the physical presence of sodium in the driver sodium-bonded spent nuclear fuel is different than that in the blanket spent nuclear fuel. Consequently, the technique and degree of difficulty for its removal depends on the type of the fuel. The EIS describes these techniques in Section 2.3.9.

14-5: As discussed in Section 2.6, the possibility of treating driver or cladded blanket spent nuclear fuel using the SRS PUREX Process was considered and dismissed from further evaluation because of the significant design modifications that would be required at SRS.

Commentor No. 15: Anonymous

Treatment and Management of Sodium-Bonded Spent Nuclear Fuel

Draft EIS Comment Form

Please address specifically how the costs for the Aluminum Based SNF Melt&Dilute facility have been addressed in the D-EIS for the Treatment&Movement of Na-Bonded SNF. 15-1

There are several ways to provide comments on the Draft EIS for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel and these include:

- attending public hearings and giving your comments directly to DOE representatives
- returning this comment form to the registration desk at a public hearing or to the address listed below
- calling toll-free and leaving your comments: 1-877-450-6904
- faxing your comments toll-free to: 1-877-621-8288
- commenting via e-mail: sodium.fuel.eis@hq.doe.gov

Name (optional): _____

Organization: _____

Home/Organization Address (circle one): _____

_____

City: _____ State: ____ Zip Code: _____

Telephone (optional): _____

COMMENTS MUST BE POSTMARKED BY SEPTEMBER 13, 1999

Response to Commentor No. 15:

15-1: Actual costs for the treatment and management of sodium-bonded spent nuclear fuel are not part of the EIS process. The SBSNF EIS was prepared in accordance with NEPA, Council on Environmental Quality regulations on implementing NEPA (40 CFR 1500-1508), and DOE's NEPA implementation procedures (10 CFR 1021). None of these require the inclusion of a cost analysis in an EIS. As discussed in the introduction, the basic objective of this EIS is to provide the public and DOE decision-makers with a description of the reasonable alternatives for the treatment and management of DOE's sodium-bonded spent nuclear fuel and their potential environmental impacts. However, DOE has issued a separate Cost Study that analyzes and compares the costs of the alternatives analyzed in the EIS. An estimate of the costs associated with treating sodium-bonded spent nuclear fuel using the melt and dilute facility at SRS is provided in Section 2.6 and Appendix B.2 of the Cost Study.

Commentor No. 16: Patricia McCracken

To: Department of Energy

August 12, 1999

From: Patricia McCracken
413 Scotts Way
Augusta, Georgia 30909
706-738-9451
fax 706-738-0637

Re: Draft Environmental Impact Statement for the treatment and management of Sodium-bonded spent Nuclear Fuel

Questions:

How do the alternatives rank in energy consumption? 16-1

How important is the experience of the workforce in the selection process? 16-2

Who has the patents on the processes presented? 16-3

How well equipped are the facilities to handle any accidents that may relate to this treatment? What would happen if a power outage or overload occurred during the electric voltage of the electrometallurgical treatment process? 16-4 16-5

What is the agency planning to do with the filters used to capture the off-gases of the melt and dilute process? What is the name of the filters to be used and how much do they cost? 16-6 16-7

What is the anticipated treatment of the filters and how will they be categorized as a waste stream? Page S-10 of the Summary (DOE/EIS-0306D July 1999. 16-6

What happens to the salt(electrometallurgical treatment process) that is removed and solidified? What is the disposal method? What is the level of contamination of the salt that becomes sorbed into the zeolite structure? How much does the glass powder cost(and who makes it) that is added to the zeolite mixture? Current market prices? 16-8 16-9

What is the level of voltage that is required for the application described on page S-13? Does this voltage vary 16-10

Response to Commentor No. 16:

16-1: The six alternatives analyzed in this EIS use the existing infrastructure at the both INEEL and SRS sites. Section 2.4 of the EIS identifies the facilities within the sites where treatment and management of the sodium-bonded spent nuclear fuel would occur. These facilities currently exist and are operational. The site-wide infrastructure characteristics are given in Sections 3.2.2 and 3.3.2 of the EIS, including annual energy consumption at each site. The energy consumed by the facilities that would be used to treat the sodium-bonded spent nuclear fuel is a small fraction of the total energy used at each site. Furthermore, none of the technologies evaluated appears to demand significantly higher or lower energy to treat the spent nuclear fuel. Section 4.14.3 provides a discussion on the relative energy consumption associated with technologies evaluated in the EIS.

16-2: Work force experience will be one of many factors taken into consideration by DOE when it selects an alternative for the treatment and management of its sodium-bonded spent nuclear fuel. At present both ANL-W and SRS have work forces that have the experience necessary to perform any of the proposed alternatives. The potential loss of experienced personnel at ANL-W was one of the factors considered when it was decided to proceed with the EIS at this time.

16-3: ANL-W did not produce any patents during the demonstration project. However, the scientists and engineers who developed the processes used in the Electrometallurgical Treatment Research and Demonstration Project patented a number of inventions related to the processes and the process equipment. Four patents were issued to cover production of the ceramic waste forms. Four more patents were issued for electrorefiner and electrorefining process inventions related to the demonstration project. All of the patents associated with the treatment processes presented in the EIS are owned by the U.S. Government.

16-4: The management facilities identified for the treatment and management of the sodium-bonded spent nuclear fuel (see Section 2.4 of this EIS) are equipped to handle spent nuclear fuel. Each facility has a well defined, approved Safety Analysis Report that documents the equipment needed to prevent and mitigate a spectrum of accidents with a likelihood of occurrence ranging from anticipated to extremely unlikely.

16-5: A disturbance in electric power supply during electrometallurgical treatment would not cause any damage to the equipment and would not lead to accidental releases of radiation to the atmosphere. The facilities where the

Commentor No. 16: Patricia McCracken (Cont'd)

16-10 (Cont'd)

from the voltage of other uses of the facilities being considered? What is the price of the energy at each site?

16-11

How much does zeolite cost?

16-12

Page S-13 states that, "In addition to the ceramic and metal waste forms of high level radioactive waste, some low-level radioactive waste would be generated." Where will this waste be stored, treated or disposed? How low is low?

16-13

Can the low-enriched uranium ingots be blended to be more enriched? What are the radiological numbers for the ingots?

16-14

Has the Purex process been modified since 1954 as described in the EIS?

16-15

Does DOE have the technology to address the incompatibility of alloys with the SRS dissolution process as described in the EIS on page S-14?

16-16

Please indicate which one of the comments in the EIS was from SRS?

16-17

Does the Purex process produce materials that could receive some treatment at the MOX facility to be built at SRS?

16-18

Does any other EIS have a process that would also help with this waste? This would be cost savings.

16-19

16-20

The University of Missouri announced some kind of new technology regarding nuclear packaging. Does this EIS have a process for evaluating various new technologies that may be under investigation?

16-21

The section S.3.3 on S-14 is confusing as it raises the question of why you should declad and /or do sodium removal when the waste can be packaged for shipment to a repository. The cover sheet states that, "One type of spent nuclear fuel that may not be suitable for disposal in a geologic repository without treatment is the DOE-owned sodium-bonded spent nuclear fuel." What does "may not" mean?

16-22

Does the waste stream vary in components from say batch to batch? Has DOE conducted a comprehensive review of all waste? May not does not sound like something that would be

16-23

16-24

16-22

Response to Commentor No. 16: (Cont'd)

treatment would be performed are equipped with multiple electric feeders and have onsite emergency diesel generators to power the equipment needed to maintain the process in a safe condition.

16-6: The off-gas system in the melt and dilute process would capture various nuclides such as cesium, tellurium, and iodine that have boiling points below or up to 1,400 °C (2,250 °F), and would be vaporized during the heating and melting process. The vaporized nuclides would be condensed and absorbed. In addition, the process would generate small quantities of oxidized actinides (e.g., plutonium, americium) that would also be captured in the filters. Depending on the level of contamination of the filters, they will be disposed of as either low-level or high-level radioactive waste. As indicated in Section 4.7.6 of the EIS, these filters would be periodically cleaned and decontaminated. The decontamination of the filters and the absorbent used to collect the volatile nuclides would produce high-level radioactive waste to be disposed of in a DOE standardized canister. The filters have not yet been designed and built. They are expected to be adsorbent to collect the volatile and gaseous fission products. Absorbents like zeolites may be used to collect cesium. Zeolite costs approximately $10 per pound. A high-efficiency particulate air filter also would be used.

16-7: The filters have not yet been designed and built, although successful tests of filter media have been conducted; therefore, the costs for the filters have not been finalized. The actual costs for the filters that would be used during treatment and management of the sodium-bonded spent nuclear fuel are not part of the EIS scope.

16-8: As described in Section 2.3.1 and Section C.1 of Appendix C of the EIS, the salt removed from the electrorefiner would be solidified, crushed, and milled; mixed with zeolite and heated where the salt is sorbed into zeolite; mixed with glass frit; and converted into a monolithic ceramic waste in a hot isostatic press. The ceramic waste form would be expected to be disposed of as a high-level radioactive waste in a geological repository. The salt would contain almost all of the fission products, including cesium and transuranic elements from the spent nuclear fuel, and would be highly radioactive.

16-9: The glass is manufactured commercially by PEMCO. For orders on the research and development scale, it costs approximately $10 per pound. The actual costs for the glass powder that would be used during the treatment and management of sodium-bonded spent nuclear fuel are not

Commentor No. 16: Patricia McCracken (Cont'd)

stated in a document that has reached this stage of public comment? 16-22

Please explain the regulatory reasons for such a statement?

What did Argonne National Laboratory-West do with the research material? Did they produce any patents? 16-25

The EIS is vague on the subject of the reactive problems of the presence of metallic sodium; "frequently by metallic uranium, which is stated to be potentially reactive; and in some cases, highly enriched uranium." The examples of metallic sodium reacting with water to produce hydrogen gas and sodium hydroxide did not give chemical formulars for these properties or volumes as compared to what? The repository problems are vague also. Your explanations are not reasonable. 16-26

Where is the explanation or details for stating that there is some "uncertainty" surrounding the acceptability of DOE spent nuclear fuel for placement in a potential geologic repository? Please give all sources that you relied on to make that statement. 16-27

Where can we locate the Settlement Agreement and Consent Order (Idaho 1995) issued on October 17, 1995, in the actions of Public Service Co. of Colorado v. Batt, No. CV 91-0035-S-EJL (D.Id.) and United States v. Batt, No. CV 91-0054-EJL9D.Id). The EIS states that 98% of the the fuel is somehow governed by this settlement and consent order. This information should be part of the EIS. Is DOE under the consent order or free to consider other comments? One of the comments in Appendix A states: "Political decisions, such as the Idaho Settlement Agreement (which says that spent nuclear fuel must be out of Idaho by 2035) should not preclude any of the No Action Alternatives from being considered." 16-28

Does any of DOE's global partners have any of this waste and what do they do? 16-29

Did any of the comments come from DOE's global partners?

Why did you issue an EIS without some comparative costs with each process and location? 16-30

Response to Commentor No. 16 (Cont'd):

16-10: During electrorefining operations, the voltage between the electrorefiner's electrodes is maintained below 1.3 volts. The electricity for the in-cell equipment comes from 480/208 volt power supplies. The electrorefining operation has been demonstrated over the last three years. The voltages employed at the electrorefiner do not have an effect on other voltage requirements for the facility.

16-11: The price of electricity at different sites is not a discriminating feature between the alternatives. The actual costs for the energy that would be consumed during treatment and management of the sodium-bonded spent nuclear fuel are not part of the EIS scope.

16-12: Zeolite costs approximately $10 per pound.

16-13: As described in the waste management subsections of Chapter 4 of the EIS, each of the processes would generate some volume of low-level radioactive waste at INEEL. This low-level radioactive waste would be packaged in management facilities at INEEL and sent to the Waste Experimental Reduction Facility for volume reduction (e.g., compaction), and then would be disposed of at the Radioactive Waste Management Complex. Low-level radioactive waste is defined in DOE Order 435.1 and in the glossary of the EIS. As explained in Section 3.2.11.4 of the EIS, the level of contamination must be below 10 nanocuries per gram to be disposed of on site. The low-level radioactive waste generated by the electrometallurgical treatment process meets this definition.

16-14: If low-enriched uranium ingots are blended with a more highly enriched uranium metal, then the enrichment of the new ingot will be higher than the original low-enriched uranium ingots, but lower than the material with which it was blended. Conversely, if low-enriched uranium ingots are blended with a lower-enriched uranium metal, then the enrichment of the new ingot will be lower than the original low-enriched uranium ingot. The uranium ingots would contain trace contamination from some fission products and actinide elements, and would generate a radiation field of about 1 to 10 rad per hour at contact, which would require shielding and remote handling. However, DOE plans to blend down the uranium metal derived from the electrometallurgical process.

16-15: The PUREX process described in the EIS is the same as that which is currently in operation at SRS's F-Canyon. PUREX has been used since 1954 and is a well-known process. While the F-Canyon has undergone

Commentor No. 16: Patricia McCracken (Cont'd)

SRS has been chosen for several missions and it would seem that cumulative activity would be important for solving unknown problems that might arise. Are two heads better than one in this science arena? — 16-31

Has this type of material ever undergone some treatment? There is some reference to remaining EBR-II spent nuclear fuel. — 16-32

How many alternatives have been dismissed and how many alternative processes are in a research stage in this science? — 16-33

In the Appendix DOE responses page A-7: Information on cost will be made available to the public via the Cost Analysis Report, which will be issued during the Draft EIS public comment period. We requested the package for the hearing in North Augusta on August 17, 1999. We did not receive the Cost Analysis Book in the DOE package. We received a cover letter and three volumes of material including DOE/EIS-0306D summary and volume I and volume II. — 16-34

Please indicate where DOE has this information? We would like to comment at the public meeting in North Augusta. We have called every number in the cover page of the material and left messages.

Who is preparing the comment books and who is responsible for printing the material for the public? Who is reading the material that is going to the public? — 16-35

Thank You to be continued...

Response to Commentor No. 16 (Cont'd):

various safety upgrades through the years, the main process itself has remained essentially unchanged.

16-16: The dissolution technology used to process spent nuclear fuel containing zirconium is well-known. A processing plant operated by Nuclear Fuel Services Inc., known as West Valley, operated from 1972 to 1978. There is also a Fluorinel Dissolution Process Facility at INEEL's INTEC facility that can process spent fuel containing zirconium. However, this facility is permanently shut down. The use of dissolution technology was considered in the list of alternatives, but was dismissed from evaluation in Section 2.6 of the EIS.

16-17: It is not clear whether the commentor is referring to technical support provided by SRS in the preparation of the EIS or public comments received from the SRS region. DOE and contractor personnel from SRS provided technical support in preparing and reviewing the EIS, especially sections that involve SRS facilities and the PUREX and melt and dilute treatment processes. Commentors on the draft EIS are identified in the comment response section of the EIS.

16-18: PUREX processing of declad and cleaned blanket spent fuel at SRS would separate plutonium from the depleted uranium and fission products in the spent fuel. The separated depleted uranium and plutonium would be stored at SRS until decisions are made about their disposition. The decision to use these materials at the mixed oxide (MOX) facility is beyond the scope of this EIS.

16-19: Some of the processes evaluated in this SBSNF EIS are also included in other EISs (e.g., the Savannah River Spent Nuclear Fuel Management EIS addresses conventional processing [PUREX], melt and dilute, and electrometallurgical treatment technologies). All potential processes have been considered for their applicability and feasibility in treating sodium-bonded spent nuclear fuel.

16-20: There is no opportunity for cost savings except for selecting the least costly treatment and management alternative in the Record of Decision. The actual costs for treating and managing sodium-bonded spent nuclear fuel are not part of the EIS process. The costs of treating and managing the sodium-bonded spent nuclear fuel are addressed in a separate Cost Study that was issued by DOE in August 1999.

Commentor No. 16: Patricia McCracken (Cont'd)

To: DOE

From: Patricia McCracken
413 Scotts Way
Augusta, Georgia 30909
706-738-9451

Re: Comments: DOE/EIS-0306D continuation of comments first sent to DOE on August 11, 1999
August 12, 1999

Page C-11 states that, "Sodium-based uranium oxide, uranium carbide, and uranium nitride fuels cannot be treated using the melt and dilute process because of their high melting points. What percentage of the material discussed pertains to these elements? 16-36

The Volume I page 248 discussion of Environmental Consequences did not discuss issues as other EIS's. For example, other DOE reports discuss "abrupt releases" that could be part of the risks. 16-37

Volume I 4.9.1 discusses an assumption of ..."sodium-bonded Fast Flux Test Facility driver spent nuclear fuels and other miscellaneous fuels assumed to be or brought to Idaho." We hope that your public meetings include all parties and locations. Exactly what are the miscellaneous fuels? No facility wants to process fuels under the title of miscellaneous. This statement is real vague in terms of worker safety. 16-38 16-39 16-38

The July SRS draft report on Paths to Closure states on page 84/Public Worker/Environmental hazards and Risks/ that, "Some declad fuel or fuel with failed cladding can unacceptably degrade current fuel storage facilities. Long-term risk develops from degradation of aluminum cladding or loss of cooling water followed by atmospheric dispersion of radioactive material. Activities planned to mitigate the risk associated with the stored fuels include: 16-40

Response to Commentor No. 16 (Cont'd):

16-21: It is assumed that the commentor is referring to ongoing research being conducted at the University of Missouri's Graduate Center for Materials Research on iron phosphate glass vitrification. This research is funded by DOE and is being conducted in collaboration with the Westinghouse Savannah River Company and Battelle Pacific Northwest National Laboratories. The purpose of this research is to develop a vitrification material for use in the treatment of nuclear waste. It is also worth noting that the University of Missouri's nuclear engineering program has been conducting research for Rockwell International Corporation on the electrochemical processing of spent nuclear oxide fuel. The purpose of this research is to determine if electrochemical processing of spent nuclear fuel could be conducted more economically than the conventional PUREX wet-chemistry process. While similar in nature to the processes evaluated in the EIS, the research being conducted at the University of Missouri does not directly support the treatment and management of sodium-bonded spent nuclear fuel. DOE evaluates new and ongoing treatment technologies on an ongoing basis. While the work at the University of Missouri has not been specifically identified in the EIS, the EIS does address the potential development of new and less mature technologies under the continued storage option of the No Action Alternative.

16-22: Section S.3.3 of the EIS Summary states that the placement of sodium-bonded spent nuclear fuel without decladding or sodium removal is considered as the direct disposal option under the No Action Alternative. The uncertain acceptability of this No Action Alternative is discussed in Section 4.12.1 of the EIS. The placement of declad and cleaned (sodium removed) blanket sodium-bonded spent nuclear fuel in high-integrity cans is considered under Alternative 2, which is described in the EIS Summary, Section S.5.3. The use of the term "may" in the cover page statement reflects the current status of the geologic repository acceptance criteria. These criteria have not been finalized and do not currently address the acceptability of placing spent nuclear fuel containing a chemically reactive material such as sodium within the repository. Until the final waste acceptance criteria are issued, it is uncertain whether spent nuclear fuel containing chemically reactive sodium would be accepted for emplacement in a geologic repository.

16-23: The waste streams can vary between batches. As part of the electrometallurgical demonstration project, waste form characterization testing has been performed on different batches to bound the performance of the waste forms. In the analyses of this EIS, it was conservatively

Commentor No. 16: Patricia McCracken (Cont'd)

Experimental Breeder Reactor II (EBR II) fuel (16.8MTHM0 to be processed in the canyons.Treatment and Storage Facility (TSF) will be constructed by 2005 for processing aluminum clad fuel (melt and dilute procedure). The processed SNF will be packaged for off site shipment. Final Al-clad SNF disposal forms will be less mobile, more manageable, and much more chemically stable; the security risk will be eliminated. It is currently assumed that transfers to the repository will begin in 2015..Stainless Steel and zircally clad SNF will be transferred to INEEL. — 16-40 (Cont'd)

How does this information correspond with the EIS for availability of storage space?

Has each proposed site worked out the permits from regulators to approve these plans in Federal Facility Agreements? If not some costs would be part of the budgets. — 16-41, 16-42

How much money was spent to develop any research to solve some of the unanswered questions of the EIS? WE noticed that the electrometallurgical treatment was an experimental demonstration project. What is the name of the report of this project and what is the contract number for reference? — 16-43, 16-44

On August 16, 1999, we did receive the Cost Study of Alternatives Presented in the Draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded spent Nuclear Fuel and a document called Nonproliferation Impacts Assessment for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel. We received this information only after great effort. — 16-45

The references of the cost study do not reflect a complete study of costs related to the alternatives.

Page S-2 of the cost summary really sums up the situation very well. Number three states:

Some of the cost estimates underlying Table S-1 are based upon conceptual designs or a partial understanding of the technical requirements for processing the spent nuclear fuel or qualifying the high-level radioactive waste products. These uncertainties are sufficiently large to make it difficult to differentiate between the cost for Alternatives 1 through 3 and Alternatives 4 and 6. — 16-46

Response to Commentor No. 16 (Cont'd):

assumed that at the time of an accident the process would contain the maximum amount of fission products within each process.

16-24: DOE has assessed, and continues to assess, the performance of waste forms that are potential candidates for disposal in a geologic repository. Waste forms from electrometallurgical treatment, the melt and dilute process, and the Defense Waste Processing Facility are included as part of that assessment.

16-25: ANL-W did not produce any patents during the demonstration project. However, the scientists and engineers who developed the processes used in the demonstration project patented a number of inventions related to the processes and the process equipment. Four patents were issued to cover production of the ceramic waste forms. Four more patents were issued for electrorefiner and electrorefining process inventions related to the demonstration project. The results of the demonstration project were published in a series of reports for DOE and the National Academy of Sciences.

16-26: The chemically reactive nature of metallic sodium is a known property. The products of such reactions are also well known and described in numerous chemistry references. Metallic uranium can react with chemicals and elements in the environment, but the unique chemically reactive feature of the spent nuclear fuel that is the subject of this EIS is its metallic sodium content. Highly enriched uranium raises a criticality concern, but it is not a unique feature of the sodium-bonded spent nuclear fuel considered in this EIS.

16-27: There is uncertainty with regard to the disposal of sodium-bonded spent nuclear fuel at this time since there are no final waste acceptance criteria for a geologic repository. DOE will be developing a final waste acceptance criteria document. The subject of waste acceptance criteria is discussed in EIS Sections 2.7 and 4.12.1. Due to the chemically reactive nature of the metallic sodium present in sodium-bonded spent nuclear fuel, its acceptability as untreated spent nuclear fuel for direct disposal currently cannot be determined. The most current version of DOE's *Waste Acceptance Systems Requirements Document* indicates that acceptable materials destined for the repository shall contain no more than trace quantities of reactive substance. Because of the chemically reactive nature of metallic sodium, it is not likely that sodium-bonded spent nuclear fuel would be acceptable in the proposed

Commentor No. 16: Patricia McCracken (Cont'd)

DOE did not do a complete study of the costs as evidenced by the references in the document. So how did they conduct their environmental assessments and dose studies on such incomplete information? 16-46 (Cont'd) 16-47

What is really confusing is the statement on S-3 of the cost report that states: " Costs for disposal of transuranic waste (which are charged incrementally) and costs for disposal of low-level radioactive waste are insignificant." The volume of wastes produced by each process and its disposal costs would seem to be significant. 16-48

If you don't have the technology then how do you begin to determine risks or environmental impacts? For example, the furnace needed for a process. 16-47

The references for cost do not show that the contractor representative called for furnace information, specifications, or other needed items for all the processes. 16-49

DOE has apparently made some agreement with Idaho that amounts to a record of decision outside the statutory regulations of the EIS. That record of decision should have been part of this EIS as a comment. Please provide all the comments in your public documents. Since this EIS is so uncertain as to risks of this waste, we do not understand any agreement that put any type of treatment or time frame for any actions by the DOE. 16-50

The idea that DOE has special environmental agreements with some states outside the regulatory NEPA policies is difficult to comprehend. A national policy of determining the Record of Decision is either in the EIS process or in the legal arena of each state. DOE's legal policies vary from state to state. We would guess without seeing the Idaho agreement that DOE did not conduct a full appeal process, thus denying the public complete representation. It does not seem legal for an EIS to be submitted to a process that has already been determined. This is fooling the public. Unless you really studied this EIS, you would not have realized that a record of decision was already determined by a DOE agreement. Considering 98% of the waste is in Idaho, this is an important issue. Furthermore, 16-51

7

Response to Commentor No. 16 (Cont'd):

16-28: The State of Idaho Settlement Agreement and Consent Order is cited in Section 1.1 of the EIS and has been added as Appendix K in the final EIS.

16-29: DOE's global partners have not used sodium-bonded nuclear fuel and have not commented on or been involved with this EIS.

16-30: The SBSNF EIS was prepared in accordance with NEPA, Council on Environmental Quality regulations on implementing NEPA (40 CFR 1500-1508), and DOE's NEPA implementation procedures (10 CFR 1021). None of these require the inclusion of a cost analysis in an EIS. As discussed in the introduction, the basic objective of this EIS is to provide the public and DOE decision-makers with a description of the reasonable alternatives for the treatment and management of DOE's sodium-bonded spent nuclear fuel and their potential environmental impacts. However, DOE has issued a separate Cost Study that analyzes and compares the costs of the alternatives analyzed in the EIS. Cost will be considered during the decision-making process in preparing the Record of Decision.

16-31: SRS was included in the SBSNF EIS preparation process. Technologies planned for or in use at SRS are part of the EIS alternatives analyzed for the treatment and management of sodium-bonded spent nuclear fuel.

16-32: As stated in the EIS, Section 1.1, some EBR-II driver and blanket sodium-bonded spent nuclear fuel assemblies have undergone electrometallurgical treatment under the research and demonstration project that has been underway at ANL-W since 1996. Also, in the 1980s 17 metric tons of heavy metal of EBR-II blanket sodium-bonded spent nuclear fuel were declad and cleaned with the U.S. Nuclear Regulatory Commission approval at the Rocketdyne facilities in California (see Section 2.3.9 in the EIS). The treated spent nuclear fuel was then shipped to SRS for further processing. It is currently stored at SRS in aluminum cans. This spent nuclear fuel is not part of the sodium-bonded spent nuclear fuel considered in this EIS, but is addressed in the Savannah River Site Spent Nuclear Fuel Management EIS.

16-33: Appendix C of the EIS describes all of the alternative treatment processes considered in the EIS. Appendix C also provides information about the maturity and the relative stage of development for each process. Section 2.6 of the EIS identifies all of the alternative treatment technologies that were considered and dismissed from detailed evaluation and the reasons for their dismissal.

Commentor No. 16: Patricia McCracken (Cont'd)

the waste of this EIS is not being presented for the miscellaneous waste by volume or by any other properties. 16-51 (Cont'd)

You need to be more specific in the terms of the EIS. For example, your glossary states that certain by-product material as defined by Section 11e(2) of the Atomic Energy Act of 1954, as amended. We would expect the contractor to define the waste in this type of scientific terms. 16-52

This EIS says that we don't know the technology but we know the risks? Just how can that be so? We haven't explained the miscellaneous waste but we know the risks. How can that be so? 16-53 16-54

The EIS can't explain most of the technology and they haven't really investigated all the costs, but they know that all these technologies are going to cost this much. How can that be so? 16-55

If DOE hasn't determined the waste form acceptance criteria then how do you know cost or risk? If you don't know timing of storage for any stage of the process, then how do you determine costs? 16-56

S-3 of the cost analysis states.."Table S-3 shows the annual costs for each alternative from 2000 to 20006, which represents the majority of the costs of the program. The time table for this chart may be very different if you consider that one process may not be ready. **Readiness of a process could be important for cost. Maybe a chart of** readiness projected timetables would help prioritize your decisions as the agency has already made decisions without the processes to carry out your legal agreements. 16-55

We believe that DOE and their contractors need to do more **extensive review of these alternatives as these cost figures and risk numbers might change with more investigation.** 16-57

The cost estimates and references did not show any consultations with electrical engineering persons regarding the voltage issue and risks. A demonstration project and actual operation might need some modifications not in this cost analysis. 16-58 16-59

Volume I of the EIS has some very positive language about environmental impacts. How can DOE say that Chaper 4 16-60

Response to Commentor No. 16 (Cont'd):

16-34: The Cost Study was issued during the public comment period, as indicated in Appendix A of the EIS. This report was mailed to interested parties on August 12, 1999, and was made available to attendees at all of the public hearings on the draft EIS

16-35: DOE is responsible for preparing this comment response document. DOE's contractors assist DOE in this task. After each comment document (e.g., letter, phone call, e-mail) is received from the public, it is read and all the comments identified within it are categorized according to their content. DOE addresses all policy-related and "out of scope" comments, while its contractors answer comments concerning technical and NEPA-related issues. As the responsible agency, DOE reviews and revises the responses to all comments, as appropriate. The completed comment response document is reviewed and approved by DOE. The Government Printing Office is responsible for printing the EIS, including the comment response document.

16-36: As indicated in Section 2.5.7 of the EIS, there are about 0.1 metric tons of heavy metal (0.2 percent) sodium-bonded spent nuclear driver fuel that is composed of uranium oxide, uranium carbide or uranium/plutonium carbide, and uranium nitride that could not be treated using the melt and dilute process. Section C.5 of Appendix C has been revised to reflect the amount of fuel that could not be treated using the melt and dilute process.

16-37: "Abrupt releases" are caused by accidents, the effects of which are analyzed in the EIS. As stated in Section 4.1 of the EIS, the evaluation of human health effects from facility accidents are presented in Appendix F. This appendix explains the methodology used to estimate the human health effects and provides descriptions of various accident scenarios, as well as the associated consequences and risks for each of the alternatives and/or management sites considered.

16-38: Fast Flux Test Facility spent nuclear fuel and other miscellaneous fuel is described in Section 2.2.3 and Appendix D, Section D.5 of the EIS. The discussion of miscellaneous fuel in Section 4.9 has been expanded to reference Appendix D for additional information.

16-39: Public hearings on the SBSNF Draft EIS were held in Idaho Falls, Idaho (August 26, 1999); Boise, Idaho (August 24, 1999); North Augusta, South Carolina (August 17, 1999); and Arlington Virginia (August 31, 1999). These [illegible] public scoping meetings were held

Commentor No. 16: Patricia McCracken (Cont'd)

provides a "detailed discussion" of the impacts on the potentially affected environmental areas when they don't even know the technology of the furnace or other environmental equipment needed for a process? 16-60 (Cont'd)

The Environmental Justice issue has not been discussed fully because the EIS does not have the complete pre-decisional report from the DOE legal group working with Idaho. For all we know the material may be deteriating rapidly and need to be moved immediately, or it may be stable until 2035. How did DOE establish the dates in the EIS for action? 16-61 16-62

"For the alternatives evaluated, DOE has determined that the proposed action would have minimal or no impacts on the remaining environmental areas (e.g., land resources, visual resources, noise, geology, and soils, ecological resources, and cultural and paleontological resources) at the proposed sites. This is because the proposed facilities already exist so, except for internal building modifications and new equipment installation, no construction activities would be required." **Since the technology is vague, and some of these alternatives indicate construction, how do you say no construction activities? What electrical equipment might be needed?** 16-63

All of the alternatives did not get a funded demonstration process. **The direct plasma Arc-Vitreous Ceramic process did not have a complete demonstration for filtration and treatment projections.** 16-64

The computer models and language used in this assessment is not reasonable and nobody can understand what all those terms not in the glossary mean. The EIS states that the "GENII computer model is well documented for **assumptions".** If the EIS has not clarified the technology then how was the computer programmed to make assumptions on unknowns? How did you calculate dose? How did they figure dose on those demonstration alternatives and on an alternative with no demonstration? 16-65 16-66

How did the computer figure finite plume air submersion options, dispersion calculations, and preliminary energy-dependent finite plume dose factors on modified equipment like furnaces with no specifications in the EIS? 16-67

Response to Commentor No. 16 (Cont'd):

aware of the public hearings, a Notice of Availability of the draft EIS was published in the Federal Register (64 FR 41404) on July 30, 1999. In addition, the public hearings were advertised in local newspapers and 1,800 post cards were sent to individuals and other interested parties.

16-40: Section 4.12.2 addresses the programmatic schedule considerations associated with alternatives involving SRS and is consistent with the current schedule of SRS activities regarding the treatment of aluminum-clad spent nuclear fuel. This EIS uses consistent assumptions regarding the use and availability of treatment and storage facilities at SRS.

16-41: Federal Facility Agreements are Agreements negotiated between DOE and EPA and/or the appropriate state regulator. These Agreements establish schedules for particular actions (i.e., compliance or cleanup activities), define responsibilities among the parties, and establish a framework for cooperation between parties. These Agreements do not contain provisions for permits. It will be noted that all facilities proposed for the treatment and management of sodium-bonded spent nuclear fuel either have or would acquire the necessary operating permits. Since there will be no substantial increase in waste generated from the treatment of sodium-bonded spent nuclear fuel, no modification to existing permits at storage and disposal facilities is necessary.

16-42: Actual costs for treating and managing the sodium-bonded spent nuclear fuel are not part of the scope of the EIS. The costs associated with obtaining any permits from regulatory agencies outside of DOE were included in the engineering cost estimate assigned to each alternative in the separate Cost Study issued in August 1999.

16-43: Figures on the total cost for the EIS (including the cost of research to address unanswered questions on the EIS) will be available after the EIS is completed and the Record of Decision is published.

16-44: The citation for the environmental assessment on the Electrometallurgical Treatment Research and Demonstration Project is provided below.

Department of Energy, 1996, "Environmental Assessment, Electrometallurgical Treatment Research and Demonstration Project in the Fuel Conditioning Facility at Argonne National Laboratory-West," DOE/EA-1148, Office of Nuclear Energy, Science and Technology, Washington, DC, May 15.

16-45: As stated in Appendix A (Table A-3), DOE committed to provide the public with a Cost Study and a Nonproliferation Impacts Assessment during the

Commentor No. 16: Patricia McCracken (Cont'd)

One alternative needed some 'new environmental equipment" just to transport. How was that factored in the computer? How did they figure contamination of filters with no scientific numbers for contamination and no disposal methods? Those conversion models could not have been accurate for radiation dose or anything else. The computer discussion did not give a list of radiation or non-radiation parameters for the reader to know what was programmed for dose.

16-68

E.3.2 Data and General Assumptions_

"To perform the dose assessments for this EIS, different types of data were collected and generated."

This section is all assumptions with no real specifications or specific data. The data cannot be generic as reported.

The statement that worker doses associated with the processing alternatives were determined from historical data associated with similar operations is not appropriate for unknown technology. The (NSRC 1999) group may have the historical data for the Purex operation but we question the numbers for a modified system. This program is oversimplified and not site specific enough for worker protection. Some of the sentences don't make sense. What does.."Thus, the only processes considered are those that are credible for the conditions under which the physical system being modeled operates." The complete specifications for these systems have not been designed so what is this language! " Although the radionuclide composition of source terms are reasonable estimates, there are uncertainties in the radionuclide inventory and release reactions that affect estimated impacts." These are not reasonable estimates because the data relied too much on estimation. We acknowledge that more data may exist than has been presented in the EIS. The references may just have been omitted.

16-69

We believe your estimated impacts should include more data based on specifications and design numbers.

There was no discussion of the various sites ability to aid workers or contain accidents (buffer zones) with each alternative. Some of these technologies may be more prone to "abrupt releases" than others.

16-70

Response to Commentor No. 16 (Cont'd):

draft EIS public comment period. The Cost Study and the Nonproliferation Impacts Assessment were mailed to interested parties on August 12, 1999, and were made available to attendees at all of the public hearings on the EIS.

16-46: During the decision-making process prior to publishing a Record of Decision, Federal agencies typically do not have detailed design information for proposed actions and alternatives. In fact, Council on Environmental Quality and DOE NEPA regulations discourage proceeding to detailed design before the NEPA process is completed and a Record of Decision is published. Cost estimates for the six alternatives and the No Action Alternative, direct disposal option, are presented in the August 1999 Cost Study and are based largely on conceptual or preliminary design information. However, cost estimates for alternatives utilizing existing spent nuclear fuel treatment facilities and/or processes (e.g., Alternative 1, electrometallurgical treatment at ANL-W and Alternative 3, PUREX at SRS) are more certain than the estimates for alternatives based on less mature technologies. Investing resources to complete detailed designs for each alternative during the NEPA review process would not be cost-effective. DOE believes the Cost Study provides the public with a reasonable comprehensive estimate of the cost of each alternative.

16-47: The EIS was prepared in accordance with Council on Environmental Quality regulations (40 CFR 1500-1508) and DOE's NEPA-related regulations (10 CFR 1021) and procedures. Environmental assessment of a new technology or a modified/enhanced version of an existing technology can be done without a complete and detailed design. In the case of a new technology, a conceptual design was used. The environmental impact analyses consider potential releases that could occur during both normal operations and accident conditions. The estimated releases were based on facility safety analysis reports. For a modified design, the environmental impacts were based on the analysis of the original design and the impacts associated with the modification were added. Both of these evaluations would be performed prior to installation and operation of the equipment. Uncertainties associated with the equipment and operation of a specific technology were captured in the evaluation by making conservative assumptions in the hazard analysis. No technology would go into service until all the requirements of the Federal and state codes and regulations were met.

16-48: The costs of disposing of the transuranic waste and the low-level radioactive

Commentor No. 16: Patricia McCracken (Cont'd)

The human health evaluation places great emphasis on dose resulting from a release and its chemical form. Yet, this EIS does not give complete chemical forms for all the alternatives and discusses miscellaneous waste. — 16-71

The use of historical doses is unclear. What time period was used for the computer models on Purex? Estimates based on similar operations would seem difficult if no other such operations existed. — 16-72

How did you do population doses for air emissions for the melt and dilute with no equipment or filter specifications? — 16-73

Some of the basic assumptions like 'Ground Surfaces were assumed to have no previous deposition of radionuclides." How did the computer programmer determine that conclusion? — 16-74

How could you determine water releases when the process would be modified from previous operations? — 16-75

The summary book page S-31 discusses the public and occupational health and safety. 'The only risk to the health and safety of the workers and the public under either option of the No Action Alternative would be from the potential exposure to radiological or hazardous chemical emissions during normal operation or accident conditions."

The risks for accidents and hazardous chemical emissions in the EIS appeared to be incomplete for the sources quoted as references E.6. We didn't even see electrical injuries listed. There was no list to determine the extent of the discussion on all potential health effects by specific chemical and site specific. The volume and changes in liquid effluent discharges would be required for state regulators and facility agreements. Facility agreements were not listed in the references. Legal agreements were not listed in the references. The EPIcode model was not site-specific and the loading of estimates and release rates on unknown parameters may not reflect a complete picture. This model is very exacting in information and not well suited to this EIS. It did not calculate all type of exposures. — 16-76

What is the complete list of chemicals that react to sodium to cause adverse impacts? What would be the release — 16-77

Response to Commentor No. 16 (Cont'd):

to the overall cost of the project, these costs contribute less than 1 percent and are insignificant in terms of discriminating between the cost of one alternative versus another.

16-49: Actual costs for treating and managing sodium-bonded spent nuclear fuel are not part of the EIS process. However, the estimates presented in the Cost Study for installing and operating furnaces were based on information from existing furnaces.

16-50: The DOE agreement with Idaho specifies that all spent nuclear fuel will be removed from Idaho by 2035. It does not specify any treatment or management alternatives for sodium-bonded spent nuclear fuel, which is approximately 2 percent of DOE's total DOE spent nuclear fuel inventory. The scope of this EIS is to evaluate and present the environmental impacts of different alternatives, as well as no action, for the treatment of one specific type of spent nuclear fuel in Idaho.

16-51: The term "miscellaneous waste" is not used in this EIS. The commentor may be referring to miscellaneous fuel, which is defined in Section 2.2.3 and Appendix D, Section D.5, of the EIS.

16-52: The definition provided in the glossary for the low-level radioactive waste is based on, and essentially equivalent to, the definition used in the Nuclear Waste Policy Act of 1982, as amended, and given in DOE Order 435.1. As stated in its accompanying manual, "[L]ow-level radioactive waste is defined by what it is not. The definition provides the framework for this concept by listing the basic radioactive waste types that are not low-level waste, thereby limiting the waste that is to be managed as low-level waste."

16-53: The EIS clearly explains the alternative technologies considered for the treatment of sodium-bonded spent nuclear fuel. Discussions of these technologies are provided in Section 2.3 and Appendix C of the EIS. Information regarding the technologies considered in the EIS is sufficient for the purposes of the EIS analysis. As explained in the response to comment 16-47, uncertainties related to equipment and technology are captured in the evaluation of impacts. These uncertainties do not prohibit and/or invalidate the evaluation of environmental impacts and the identification of the potential risks associated with each alternative.

16-54: DOE assumes the comment to be referring to "miscellaneous fuel" and not "miscellaneous waste," as stated. In response to miscellaneous fuel, the EIS has clearly identified the elements of this fuel category in

Commentor No. 16: Patricia McCracken (Cont'd)

rates for each of the substances released? For example, the EIS mentions wet storage rupture for SRS. 16-77 (Cont'd)

DOE may need to update its dialogue with proposed sites as persons referenced in the EIS may no longer be in that program. Also DOE contractor persons apparently have changed jobs. 16-78

Thank you for the opportunity to comment.

Response to Commentor No. 16 (Cont'd):

Section 2.2.3 and Appendix D.5, as explained in the response to comment 16-38. This fuel category was considered to be driver spent fuel type, and its risks were evaluated in the EIS.

16-55: The Cost Study was based on an extrapolation of historical costs for comparable operations. The cost for waste form qualification is consistent with other experiences and assumptions within the DOE complex. Uncertainties in the maturity of the technologies are accounted for by the contingency factors used in the Cost Study, with less mature technologies requiring a higher contingency factor. The Cost Study incorporates schedule considerations for each alternative. Estimating the actual costs for treating and managing sodium-bonded spent nuclear fuel is not part of the scope of the EIS.

16-56: As stated in the introduction to the EIS, the programmatic risk in implementing any of the potential alternatives for the treatment and management of sodium-bonded spent nuclear fuel, or of not treating this fuel, is the uncertainty surrounding the acceptability of DOE spent nuclear fuel for placement in a potential geologic repository. Although not final, the latest guidance provided by DOE's Office of Civilian Waste Management in their "Waste Acceptance System Requirements Document," Revision 3, April 1999 (see Section 4.12.1 of the EIS), indicates that it is highly probable that sodium-bonded spent nuclear fuel would not be acceptable in the repository without some stabilization and/or removal of the metallic sodium. The stabilization of the spent nuclear fuel and/or removal of the metallic sodium will provide greater protection of human health and the environment.

16-57: During the decision-making process prior to publishing a Record of Decision, Federal agencies typically do not have detailed design information for proposed actions and alternatives. In fact, Council on Environmental Quality and DOE NEPA regulations discourage proceeding to detailed design before the NEPA process is completed and a Record of Decision is published. Cost estimates for the six alternatives and the No Action Alternative, direct disposal option, are presented in the August 1999 Cost Study and are based largely on conceptual or preliminary design information. However, cost estimates for alternatives utilizing existing spent nuclear fuel treatment facilities and/or processes (e.g., Alternative 1, electrometallurgical treatment at ANL-W and Alternative 3, PUREX at SRS) are more certain than the estimates for alternatives based on less mature technologies. Investing resources to complete detailed designs for each alternative during the NEPA

provides the public with a reasonable comprehensive estimate of the cost of each alternative.

16-58: A number of electrical engineers and industrial safety engineers were involved in the design, installation, and qualification of the equipment used during the electrometallurgical demonstration project. The costs associated with the demonstration project were used as the basis for estimating the cost of electrometallurgical treatment in the Cost Study. The risks from electrometallurgical treatment related to voltage are small (see response to comment 16-11).

16-59: Such modifications are anticipated. They are taken into account in the Cost Study through contingency factors.

16-60: The uncertainties associated with the development and testing of a new furnace for the melt and dilute treatment process would require a demonstration project that would delay process readiness and implementation. Any technical uncertainties would be resolved before the start of operation. The environmental impacts associated with operation of the furnace, which is an electric induction furnace, for the melt and dilute process were estimated consistent with the methodology described in response to comment 16-47 above.

16-61: It is not clear what predecisional report the commentor is citing. DOE is committed to full compliance with all provisions of Executive Order 12898. The environmental justice analysis was prepared in compliance with the Council on Environmental Quality's guidelines for inclusion of environmental justice under NEPA. The EIS addresses the issue of whether implementation of the proposed action or alternatives would result in disproportionately high and adverse environmental effects on minority populations or low-income populations. The Council's guidance further states that an environmental effect must be significant to qualify as disproportionately high and adverse. The term "significant" is discussed in the Council's implementation regulations (see 40 CFR 1508.27 and Appendix H, Section H.2 of this EIS). As discussed in Chapter 4 of the EIS, implementation of the alternatives for the treatment and management of sodium-bonded spent nuclear fuel would pose no significant radiological or nonradiological health risks to the public. The maximum estimated incremental dose to an average individual from the treatment and management of sodium-bonded spent nuclear fuel would be approximately 0.05 percent of natural background radiation. These risks would not be significant

Commentor No. 16: Patricia McCracken

Response to Commentor No. 16 (Cont'd):

regardless of the racial, ethnic, and economic composition of the potentially affected populations.

16-62: In accordance with DOE's Programmatic Spent Nuclear Fuel EIS, the No Action Alternative for this EIS assumes that each sodium-bonded spent nuclear fuel assembly is examined for integrity (i.e., stabilization activities) before it is placed in storage. Dates in the EIS are based on the availability of facilities and treatment time for each alternative and technology.

16-63: Section 4.1 of the EIS further explains why impacts to land resources, visual resources, noise, geology, soils, ecological resources, and cultural and paleontological resources will not occur. It should be noted that, although some of the technologies are less well developed than others, enough is known about them to indicate that only internal equipment modifications are needed. Current electrical equipment is expected to be adequate to meet project demands.

16-64: Each potential sodium-bonded spent nuclear fuel treatment technology was evaluated based on current knowledge and experience with that technology. The direct plasma arc-vitreous ceramic process was considered in the EIS and, as discussed in Section 2.6 of the EIS, was dismissed for further evaluation. Not all of the technologies analyzed have had a complete demonstration project.

16-65: The SBSNF EIS was prepared in accordance with Council on Environmental Quality regulations (40 CFR 1500-1508) and DOE's NEPA-related regulations (10 CFR 1021) and procedures. The computer codes used in the preparation of this EIS are well documented for assumptions, technical approach, methodology, and quality assurance issues. These codes have been subjected to extensive quality assurance and quality control, including a comparison of the results from the model computations with those from hand calculations and the performance of internal and external peer reviews.

16-66: The GENII computer program that was used to estimate the human health effects from releases of radioactive material during normal operation and accidental conditions is a well-known program, and its applicability has been demonstrated in various DOE EISs. The program models the dispersion of releases and calculates potential doses to the public and individuals residing in the vicinity of the facility. All required input to this program is well defined and the process is well understood. The evaluation is

during normal and accident conditions. As explained in the response to comment 16-47, the releases were estimated based on facility safety analysis reports.

16-67: Atmospheric dispersion of radioactive material releases vary depending on the type and duration of the release. The selection of a dispersion model is an input to the GENII computer program. The dispersion models used in the program are well defined and are explained in the program manual. These models are independent of the technologies used.

16-68: The expression "new environmental equipment" is not used in the EIS and new environmental equipment is not related to the use of a computer program. Contamination in the off-gas system filters originates from the process. Each process is well defined. For example, because of the high temperature used in the melt and dilute process, some radionuclide elements with boiling temperatures below the process temperature would evaporate. Some elements would be oxidized and released to the off-gas system. The gaseous flow through the off-gas system first would be condensed and adsorbed, and then filtered before entering the atmosphere. All noble gases would pass through the filters, but only a small fraction of particulates would pass through filters. The specific assumptions on various filtration factors are given in Appendix E and Appendix F. These appendices also provide the source terms associated with each of the releases considered.

16-69: Appendix E, Section E.3.2, of the EIS provides the data and general assumptions for both generic and site-specific data. Clarifications have been added to each data category to differentiate between the generic and site-specific data. For example, meteorological, population, and source terms data are all site-specific, whereas annual exposure time to plume and ground contaminations are generic data. The estimated worker dose under each alternative is given in Section E.4 of Appendix E. EIS preparers used a standard approach for estimating average and total worker doses that is based on doses received during similar activities within each management facility. The text describing the analysis of uncertainties has been revised for clarification and is applicable to the spent nuclear fuel processed under this EIS.

16-70: Facility and site emergency procedures for accident conditions are included in the operational procedure manual and are documented in the facility Safety Analysis Report. The facility Safety Analysis Report identifies and analyzes the various accident scenarios that could occur during operation and determines their consequences to the public. The operation

Commentor No. 16: Patricia McCracken

Response to Commentor No. 16 (Cont'd):

of a new technology would start only after the facility has met all required regulations, including those that protect the worker and general public. Appendix F of the EIS evaluates a spectrum of accidents that could occur during the treatment process, and also estimates the human health effects associated with each of the accidental radiological and chemical material releases.

16-71: The severity of internal exposure from radiation sources entering the human body through either inhalation or ingestion depends on the chemical form (solubility) of the radioactive material. The analysis in this EIS assumes the worst case solubility scenario, which results in the maximum dose. This is an input parameter to the GENII code. The use of the worst case solubility scenario was added to the list of basic assumptions in Section E.3.2 of Appendix E.

16-72: ANL-W worker doses were estimated based on historical data associated with similar activities. No computer modeling was used to estimate such doses. Similar activities are not necessarily identical activities. For example, electrometallurgical treatment activities include fuel handling activities (i.e, retrieving, dismantling, assembling, transporting) that were performed at ANL-W during experimental breeder reactor operation. Almost all of these activities would occur in a hot cell with remote operation (robotic) tools. Historical dose data on these activities can be used to estimate the worker dose. The average SRS worker dose used to evaluate environmental impacts is routinely assumed to be 500 millirem per year. This dose value is conservative and has been published in numerous EISs. As indicated in Section E.4.3 of Appendix E of this EIS, this average SRS worker dose estimate was used in the SRS Spent Nuclear Fuel Management EIS to estimate the impact of activities similar to those described in this SBSNF EIS.

16-73: It is standard practice to install one or more banks of high-efficiency particulate air filters, known as high efficiency particulate air filters, in the off-gas system. Filter specifications would not be needed to evaluate environmental impacts. Each bank of high efficiency particulate air filters would absorb at least 99.9 percent of the particulates. The use of two banks of filters would reduce the particulate release to the atmosphere by a factor of 1 million from that generated in the process. Only gaseous fission products such as krypton, iodine, and tritium would pass through high efficiency particulate air filters without being absorbed. The iodine gases would be

filters. At least 99 percent of iodine would be absorbed in a bed of charcoal filters. The off-gas system exhaust would enter the facility exhaust system and would pass through another bank of high efficiency particulate air filters. Therefore, a very small fraction (one in a billion) of particulates generated in the melt and dilute process would be released to the environment.

16-74: The analysis in this EIS determined the incremental heath effects associated with the implementation of each alternative. Previously contaminated ground is part of the baseline dose, which is independent of the health effects associated with operation of any one of the treatment processes. Baseline doses to the public at each of the management sites are given in Chapter 3 of the EIS.

16-75: A modification to a process would identify potential changes to a liquid or gaseous effluent. Therefore, for the purposes of environmental impact evaluation, it is known whether a modification would lead to liquid effluent releases.

16-76: For each alternative, the EIS summarizes the risks from releases of hazardous chemicals during both normal operation and accident conditions. Discussions of risk in Chapter 4 are cross-referenced to Appendices E and/or F for further details. For example, under Alternative 1, Section 4.3.4.2 provides the consequences of accidents involving hazardous chemicals in Table 4-17, with a reference to Section F.3.1.2 of Appendix F for details. The chemicals involved in these accidents were uranium and cadmium. Appendix E, Section E.6, lists the references used in that appendix. As indicated, the Savannah River Spent Fuel Management EIS was the source for information about chemical releases during normal operation at SRS. Electrical injuries are considered industrial accidents and are not expected to be affected by any of the alternatives evaluated in this EIS. For example, electrical equipment used in the electrometallurgical treatment process, which has been in operation for over three years, is located in a hot cell (remotely operated); no electrical injuries are expected to result from the remote operation of this equipment. Every operation under the proposed action would be carried out under procedural and operational controls. With regard to permits and regulatory/facility agreements, Chapter 3 of the EIS provides the baseline conditions at each site and lists the applicable standards and/or regulations in each of the resources described. Since there would be no new construction as a result of the proposed action, no regulation and/or standard would be affected. As explained in various sections of Chapter 4 of the EIS, the volume and changes in the effluent discharges would be

Commentor No. 16: Patricia McCracken

Response to Commentor No. 16 (Cont'd):

within the applicable permits and standards. With regards to analysis using the EPIcode™, the only input that was not site- and accident-specific was meteorology. The code does not have the capability to use site meteorology data and is limited to a specific condition (e.g., stability and wind speed). The calculations in this EIS and the applicability of the EPIcode™ and its characteristics are based on a conservative meteorological condition. The applicability of the EPIcode™ and its characteristics are described in Appendix F, Section F.3.1.1. The methodology used to estimate accidental releases of hazardous chemicals also is discussed in Appendix F. In addition, see the responses to comments 16-47, 16-61, and 16-37.

16-77: Openly available chemical references provide details on the nature of chemical reactions with sodium. The release rates for each substance are not relevant to this EIS because the fact that metallic sodium reacts with air and water to produce hydrogen is sufficient to characterize the sodium as chemically reactive and potentially unstable in a geologic repository environment. Current storage conditions for sodium-bonded spent nuclear fuel are monitored. Some sodium-bonded spent nuclear fuel is currently in wet storage at INEEL, not SRS. Some wet storage container leakage has been inferred by the presence of bubbles on the containers, but no dangerous conditions have been found. This EIS does not mention wet storage rupture at SRS.

16-78: DOE Headquarters staff has maintained a dialogue with the site personnel working on the EIS throughout the preparation of the document to ensure that all information is as accurate and up-to-date as possible. Chapter 7 of the EIS accurately reflects the personnel who worked on this EIS.

Commentor No. 17: Steve Hopkins

Notes from Steve Hopkins
8/24/99 - Boise, Idaho

Pyroprocessing raises significant proliferation risks. A National Academy of Sciences (NAS) report commissioned by DOE explained that the process "could be redirected to produce material with nuclear detonation capability." The report also raised questions about interim storage of the waste streams and other aspects of pyroprocessing.

...with some modifications, plutonium could be produced..." James Warf

"Probably the greatest hazard arises from spreading sophisticated technologies around the world, technologies which make reprocessing spent fuel easier and possible in facilities small enough to conceal underground." **17-1**

In 1994, DOE secretary Hazel O'Leary asked Congress to stop funding the IFR. "Because it is based on plutonium reprocessing and recycle, continued development of the Integral Fast Reactor would undercut our efforts to discourage other countries from plutonium reprocessing and recycle."

A DOE source quoted in an industry trade journal (Inside Energy) said that at Argonne-West, pyroprocessing is "just about the only thing they have left to do...It's a jobs issue." Nucleonics Week, June 8, 1995. **17-2**

A 1996 NAS study: "could be used by another country to obtain plutonium for a weapons program."

Another NAS study:
"Although the developers of the electra...technique argue that the technology is proliferation resistant, any SNF processing approach that is capable of separating fissionable materials from associated fission products and transuranic elements could be redirected to produce material with nuclear detonation capability...Demonstration of the process could, however, add to the risk that a nation intent on weapons production might consider adapting this technology for possible production of fissile material, although such material would be of poor quality for a weapon. **17-1**

Response to Commentor No. 17:

17-1: Assessment of nonproliferation impacts is not a part of the scope of the EIS. However, DOE's Office of Arms Control and Nonproliferation assessed the potential nonproliferation impacts that may result from each of the alternatives and technologies analyzed in this EIS. This Nonproliferation Impacts Assessment stated that, for this specific application, electrometallurgical treatment is acceptable in terms of nonproliferation risk.

17-2: ANL-W is involved in other DOE missions in addition to electrometallurgical treatment. Ongoing activities unrelated to electrometallurgical treatment at ANL-W include long-term waste storage gas generation testing at the Zero Physics Power Reactor; characterization and repackaging of mixed hazardous waste for shipment to the Waste Isolation Pilot Project at the Hot Fuel Examination Facility; conversion of sodium coolant from the EBR-II and Fermi reactors to chemically inert low-level radioactive waste in the sodium process facility; and deactivation of the EBR-II facility. The number of jobs affected by the electrometallurgical treatment alternative at ANL-W is presented in Section 4.2.3 of the EIS.

Commentor No. 18: Anonymous

Treatment and Management of Sodium-Bonded Spent Nuclear Fuel

Draft EIS Comment Form

I SUPPORT THE PLAN AS OUTLINED. THIS SHOULD BE DONE AT INEL-W Electrometallurgical Treatment option

18-1

There are several ways to provide comments on the Draft EIS for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel and these include:

- attending public hearings and giving your comments directly to DOE representatives
- returning this comment form to the registration desk at a public hearing or to the address listed below
- calling toll-free and leaving your comments: 1-877-450-6904
- faxing your comments toll-free to: 1-877-621-8288
- commenting via e-mail: sodium.fuel.eis@hq.doe.gov

Name (optional): _____

Organization: _____

Home/Organization Address (circle one): _____

_____

City: _____ State: ____ Zip Code: _____

Telephone (optional): _____

COMMENTS MUST BE POSTMARKED BY SEPTEMBER 13, 1999

Response to Commentor No. 18:

18-1: The commentor's support for the electrometallurgical treatment of sodium-bonded spent nuclear fuel at ANL-W is noted.

Commentor No. 19: Anonymous

Draft EIS Comment Form

Treatment and Management of Sodium-Bonded Spent Nuclear Fuel

1. Why ARE you making off-gas filters in HLW? The can be treated and disposed of ASTLW per all of the allowable DOE Orders. — 19-1

2. Where is the documentation to prove this fuel is RCRA reactive? — 19-2

There are several ways to provide comments on the Draft EIS for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel and these include:

- attending public hearings and giving your comments directly to DOE representatives
- returning this comment form to the registration desk at a public hearing or to the address listed below
- calling toll-free and leaving your comments: 1-877-450-6904
- faxing your comments toll-free to: 1-877-621-8288
- commenting via e-mail: sodium.fuel.eis@hq.doe.gov

Name (optional): _____

Organization: _____

Home/Organization Address (circle one): _____

_____

City: _____ State: _____ Zip Code: _____

Telephone (optional): _____

COMMENTS MUST BE POSTMARKED BY SEPTEMBER 13, 1999

For more information contact: Susan Lesica, NE-40
U.S. Department of Energy • 19901 Germantown Road • Germantown, MD 20874
Toll-free Telephone: 1-877-450-6904 • Toll-free Fax: 1-877-621-8288
E-mail: sodium.fuel.eis@hq.doe.gov • Website: http://www.ne.doe.gov/home/eis.html

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06/10/99

Response to Commentor No. 19:

19-1: The off-gas system in the melt and dilute process would capture various nuclides such as cesium, tellurium, and iodine that have boiling points below up to 1,400 °C (2,250 °F) and would be vaporized during the heating and melting process. The vaporized nuclides would be condensed and absorbed. In addition, the process would generate small quantities of oxidized actinides (e.g., plutonium, americium) that also would be captured in the filters. Depending on the level of contamination of the filters, they will be disposed of as either low-level or high-level radioactive waste. As indicated in Section 4.7.6 of the EIS, these filters would be periodically cleaned and decontaminated. The decontamination of the filters and the absorbent used to collect the volatile nuclides would produce high-level radioactive waste to be disposed of in a DOE standardized canister.

19-2: Metallic sodium reacts vigorously with water or moist air to produce heat, potentially explosive hydrogen gas, and sodium hydroxide, a corrosive substance. One of the primary goals of RCRA is to ensure that waste is managed in an environmentally sound manner. As discussed in Section 4.12.1 of the EIS, untreated sodium-bonded spent nuclear fuel may be regulated by RCRA, since it exhibits certain characteristics considered hazardous; that is, it is ignitable as defined in 40 CFR 261.21, corrosive as defined in 40 CFR 261.22, and reactive as defined in 40 CFR 261.23. However, this determination has not been made. Thus, the presence of metallic sodium could complicate qualification of this spent nuclear fuel for ultimate disposal in a geologic repository.

Commentor No. 20: Beth Duke

Forward Header

Subject: Draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel
Author: beth@sunvalleymag.com_at_INTERNET
Date: 8/23/99 11:34 AM

August 22, 1999

As a citizen of the United States, and a resident of Idaho, I would like to make the following comments on the Draft Environmental Impact Statement for the Treatment and Management of Sodium Bonded-Spent Nuclear Fuel.

First, I would ask that you extend the public comment period at least 60 days, since much information that is relevant to the procedure will not be available until after the current comment period is closed. For the public to be able to accurately access this draft EIS, and to make accurate comments, this is the very least the Department of Energy should do. 20-1 20-2

Secondly, I would like to go on record as objecting to pyroprocessing by the INEEL for the following reasons: 20-3

1. The entire concept runs counter to the US nonproliferation goals since it separates out bomb-grade uranium from spent fuel and the technology can be modified to separate out bomb-grade plutonium. 20-4

2. Take taxpayer dollars away from greater environmental problems at the INEEL. 20-5

3. Creates new forms of nuclear waste. 20-6

4. Wastes taxpayer dollars (as emphasized on NBC's "Fleecing of America). 20-5

Thank you,

Beth M. Duke
PO Box 964
Sun Valley, Idaho 83353
788-0770 (work)

Response to Commentor No. 20:

20-1: In an effort to ensure that all interested parties had time to comment on the draft EIS, the deadline for transmittal of comments was extended from September 13 to September 28, 1999 (64 FR 49169).

20-2: DOE made materials supporting preparation of the EIS available in the public reading rooms and at the public hearings held on August 17, 1999, in North Augusta, South Carolina; August 24, 1999, in Boise, Idaho; August 26, 1999, in Idaho Falls, Idaho; and August 31, 1999, in Arlington, Virginia. These materials included the environmental assessment for the Electrometallurgical Treatment Research and Demonstration Project, the Finding of No Significant Impact for the environmental assessment, National Research Council reports, the 1995 Settlement Agreement and Consent Order with the State of Idaho, scoping period meeting transcripts and comments, and the draft EIS hearing presentations and fact sheets. In addition, completion of the Cost Study and Nonproliferation Impacts Assessment was expedited so that they also would be available to the public during the comment period. These reports were mailed to interested parties on August 12, 1999, and were available at the public hearings on the draft EIS. Although these reports are not critical to the environmental impact analysis presented in the EIS, they will be considered during the decision-making process in the preparation of the Record of Decision. While the final National Research Council report on the Electrometallurgical Treatment Research and Demonstration Project at ANL-W was published April 2000, interim status reports were produced throughout the project. Data generated during the demonstration project were used in preparing the EIS, as discussed in Section 1.6.3 of the EIS.

20-3: The commentor's objection to electrometallurgical treatment (pyroprocessing) of sodium-bonded spent nuclear fuel at INEEL is noted.

20-4: The assessment of nonproliferation impacts is not a part of the EIS process. The Nonproliferation Impacts Assessment stated that electrometallurgical treatment, for this specific application, would not result in an increase in weapons-usable fissile material inventories. Although highly enriched uranium would be an interim product of electrometallurgical treatment, it would be downblended to low-enriched uranium during treatment. Within the current equipment configuration and design, it is not possible to produce weapons-usable plutonium by adjusting operating parameters. Traditional aqueous processing would have to be used after electrometallurgical treatment. However, traditional aqueous processing could also be used to produce weapons-usable plutonium directly from the spent nuclear fuel

Commentor No. 20: Beth Duke

Response to Commentor No. 20 (Cont'd):

20-5: Congress determines how funds are allocated. DOE spends monies consistent with Congressional direction. DOE is not in a position to make the difficult tradeoffs that may be required between alternative Federal programs and spending priorities. The issue of funding for the treatment and management of sodium-bonded spent nuclear fuel is beyond the scope of the SBSNF EIS.

20-6: All of the alternatives evaluated in this EIS would produce some forms of high-level radioactive waste. Electrometallurgical treatment (or pyroprocessing) would produce two new waste forms, both of which are more stable than untreated sodium-bonded spent nuclear fuel. DOE expects that these waste forms would be suitable for disposal in a geologic repository.

Commentor No. 21: Bpdufur@micron.net

Forward Header

Subject: Draft Environmental Impact Statement of the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel
Author: bpdufur@micron.net at INTERNET
Date: 8/25/99 9:33 PM

I am commenting on the Draft Environmental Impact Statement of the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel.

Please extend the comment period for the above subject. The study done by the National Academy of Sciences of the proposed treatment needs to be honestly reviewed. 21-1 21-2

Response to Commentor No. 21:

21-1: In an effort to ensure that all interested parties had time to comment on the draft EIS, the deadline for transmittal of comments was extended from September 13 to September 28, 1999 (64 FR 49169).

21-2: The National Academy of Sciences' National Research Council prepared interim status reports on the results of the Electrometallurgical Treatment Research and Demonstration Project that have been reviewed by DOE. The National Research Council completed their evaluation of the electrometallurgical treatment demonstration project in September 1999 and published their final summary report in April 2000. The final report findings will be considered during the decision-making process leading to the Record of Decision.

Commentor No. 22: Doug Turner

Forward Header

Subject: Fwd:Comments on Draft EIS for Na Bonded SNF
Author: <EMTEIS@hq.doe.gov>
Date: 9/16/99 8:32 AM

Forward Header

Subject: Comments on Draft EIS for Na Bonded SNF
Author: dwz@ornl.gov_at_INTERNET
Date: 9/15/99 11:47 AM

Comments on the Draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel
Section S.2 of the EIS describes the types of sodium bonded fuel in the DOE inventory in terms of five distinct categories: EBR-II Driver, EBR-II Blanket, Fermi-1 Blanket, Flast Flux Test Facility Driver, and Miscellaneous. The Miscellaneous category includes small lots of sodium bonded SNF at various sites (Hanford, Oak Ridge, Savanah River, and Sandia) which are to be shipped to INEEL per the record of decision on the PEIS for SNF. However, section S.5 of the EIS describes the six evaluated alternatives (which are combinations of No Action, Electrometallurgical Treatment at ANL-W, PUREX at SRS, and Melt & Dilute at SRS or ANL-W) solely in terms of application for either blanket fuel or driver fuel categories. 22-1

There is no mention of the Miscellaneous category in the evaluated alternatives. If this EIS is intended to address the disposition of the Miscellaneous sodium bonded SNF category, the relationship between the Miscellaneous category and the driver and blanket fuel disposition alternatives should be explicitly stated.

If there are any questions about these comments, please contact either me or Brian Oakley at 423-241-3061.

Doug Turner
Bechtel Jacobs Company LLC
7078F, MS 6402
ph 423-576-2017; fax 423-241-5049
pager 873-5378; dwz@ornl.gov

Response to Commentor No. 22:

22-1: The miscellaneous sodium-bonded spent nuclear fuel is described in Appendix D, Section D.5. For the purposes of this EIS, all miscellaneous sodium-bonded spent nuclear fuel is considered to be driver fuel. Section 2.2.3 of the EIS has been revised to provide this clarification.

Commentor No. 23: Matt Smith

From: msmith@computer-depot.com_at_INTERNET at X400PO
Date: 8/22/99 11:12AM -0700
To: EMTEIS at NE-02
*cc: SSmith7235@aol.com_at_INTERNET at X400PO
Subject: Pyroprossing

I am writing you in regard to the proposed Pyroprocessing program At INEEL. 23-1

There must be a 60 day extension of the comment period in order to adequately address all the concerns.

It appears the DOE is missing 23-2

1. The demonstration project results on pyroprocessing. 23-3
2. A National Academy of Sciences review of the project. 23-4
3. Cost analysis of the various alternatives. 23-5
4. A nuclear weapons proliferation assessment by the Department of Energy.
5. Yucca Mountain, the purported destination of the waste, has not done a detailed environmental impact study on accepting the waste. The history to date indicates that neither Yucca Mountain or WIIP will accept this type of waste. 23-6

Once again Idaho with a volatile record of earth quakes in this area and an acquifer that is world renown will be forced to hold indefinitely this dangerous material along with our breath in the hopes that the predictions of future eruptions will not occur. This is highly unacceptable with no more assurances and concrete Evidence that we have at the present time that we are not at risk. 23-7

Yours truly,

Matt Smith

Response to Commentor No. 23:

23-1: In an effort to ensure that all interested parties had time to comment on the draft EIS, the deadline for transmittal of comments was extended from September 13 to September 28, 1999 (64 FR 49169).

23-2: The Electrometallurgical Treatment Research and Demonstration Project began in June 1996 and, although the test results have not yet been finalized in a single report, a number of status reports issued by the National Academy of Sciences' National Research Council Committee were considered in the preparation of the draft EIS. The success criteria established at the outset of the project were fulfilled. The environmental impact analysis associated with the electrometallurgical treatment process alternatives was based on actual data from the project. Section 1.6.3 of the EIS summarizes the status and the results of the demonstration project.

23-3: The National Academy of Sciences' National Research Council prepared interim status reports on the results of the Electrometallurgical Treatment Research and Demonstration Project that have been reviewed by DOE. The National Research Council completed their evaluation of the electrometallurgical treatment demonstration project in September 1999 and published their final summary report in April 2000. The final report findings will be considered during the decision-making process leading to the Record of Decision.

23-4: The SBSNF EIS was prepared in accordance with NEPA, Council on Environmental Quality regulations on implementing NEPA (40 CFR 1500-1508), and DOE's NEPA implementation procedures (10 CFR 1021). None of these require the inclusion of a cost analysis in an EIS. As discussed in the introduction, the basic objective of this EIS is to provide the public and DOE decision-makers with a description of the reasonable alternatives for the treatment and management of DOE's sodium-bonded spent nuclear fuel and their potential environmental impacts. However, DOE issued a separate Cost Study on August 12, 1999, that analyzes and compares the costs of the alternatives analyzed in the EIS. Cost will be considered during the decision-making process in preparing the Record of Decision.

23-5: Although the assessment of nonproliferation impacts is not a part of the EIS process, DOE's Nonproliferation Impacts Assessment was mailed out to interested members of the public on August 12, 1999, and is available by request. The assessment was also placed in the DOE public reading rooms and distributed at the public hearings held during the public comment

Commentor No. 23: Matt Smith

Response to Commentor No. 23 (Cont'd):

risk will be considered during the decision-making process leading to the Record of Decision.

23-6: This SBSNF EIS does not specify a site for an ultimate geologic repository. Only preliminary waste acceptance criteria currently exist. Conclusions regarding the acceptability of the different waste forms for each alternative are addressed in the EIS. As discussed in Section 1.6.2.2 of this EIS, the Yucca Mountain Draft EIS has been issued by DOE. The draft waste acceptance criteria for Yucca Mountain currently only address defense waste processing facility high-level waste logs and commercial spent nuclear fuel as acceptable. DOE expects that the waste products described for all the alternatives analyzed in detail in the SBSNF EIS will be acceptable in the final waste acceptance criteria for Yucca Mountain.

23-7: As a result of its agreement with the State of Idaho, DOE is developing a treatment process to facilitate the disposal of the sodium-bonded spent nuclear fuel. Under this agreement, all spent nuclear fuel will be moved out of Idaho by the year 2035. The alternatives analyzed in this EIS treat the sodium-bonded nuclear fuel and create waste forms that would most likely be acceptable for disposition in a geologic repository. As described in Chapter 4 of this EIS, under all alternatives no radiological liquid effluent would be discharged to the groundwater or the aquifer at the INEEL site. Evaluations of the radiological impacts associated with an earthquake have shown the risk of latent cancer fatalities to a member of the public residing within 80 kilometers (50 miles) of the site to be much lower than 1. Therefore, as a result of the proposed action, no measurable increase in the number of latent cancer fatalities in the surrounding population is expected for a postulated earthquake in the INEEL area.

Commentor No. 24: Monte Wilson

TO: Susan Lesica, USDOE
FROM: Monte D. Wilson, 1055 Dobyns Lane, Potlatch, ID 83855
DATE: August 29, 1999
RE: Comments on the Draft EIS for Treatment and Management of Sodium-Bonded SNF

Inasmuch as the draft EIS places primary emphasis on Electrometallurgical technology, I request that the comment period be extended until: 24-1

a) the electrometallurgical treatment demonstration project has been completed and thoroughly evaluated, and b) the National Academy of Sciences completes its review of the electrometallurgical treatment process. 24-2, 24-3

I recommend that the two separation technologies under consideration (EMT and PUREX) be rejected because they would be in conflict with US nonproliferation goals and because they would produce multiple, and new waste forms. 24-4, 24-5

I recommend that a non-separation technology be used for Treatment of all sodium-bonded spent nuclear fuel.

I recommend that the High Integrity Cans be used for disposal of The blanket fuel. 24-6

I recommend that further development work be done to determine if it is feasible to remove sodium from the driver fuel and then dispose of the driver fuel in High Integrity Cans. If the process is feasible, I recommend that it be utilized for disposal of all driver fuel.

If the sodium removal and disposal in High Integrity Cans is Ultimately shown to be not feasible for the driver fuel, then I recommend that it be prepared for disposal by some other non-separation technology such as the Glass Material Oxidation and Dissolution System (GMODS) or the Direct Plasma Arc-Vitreous Ceramic Treatment process. 24-7

Response to Commentor No. 24 :

24-1: In an effort to ensure that all interested parties had time to comment on the draft EIS, the deadline for transmittal of comments was extended from September 13 to September 28, 1999 (64 FR 49169).

24-2: The draft EIS did not emphasize the electrometallurgical treatment technology over the other process technologies. The Electrometallurgical Treatment Research and Demonstration Project began in June 1996 and, although the test results have not been finalized in a single report, a number of status reports issued by the National Academy of Sciences' National Research Council Committee were considered in the preparation of the draft EIS. The success criteria established at the outset of the project were fulfilled. The environmental impact analysis associated with the electrometallurgical process alternatives was based on actual data from the project. Section 1.6.3 of the EIS summarizes the status and the results of the project.

24-3: The National Academy of Sciences' National Research Council prepared interim status reports on the results of the Electrometallurgical Treatment Research and Demonstration Project that have been reviewed by DOE. The National Research Council completed their evaluation of the electrometallurgical treatment demonstration project in September 1999 and published their final summary report in April 2000. The final report findings will be considered during the decision-making process leading to the Record of Decision.

24-4: The commentor's opposition to electrometallurgical treatment and PUREX is noted. DOE is concerned with the nonproliferation impacts of all its proposed actions, although the assessment of nonproliferation impacts is not a part of the EIS process. For this reason a separate Nonproliferation Impacts Assessment was prepared by DOE's Office of Arms Control and Nonproliferation. This assessment stated that, for this specific application, all alternatives except PUREX processing at SRS are fully consistent with U.S. policy concerning reprocessing and nonproliferation. Information from this assessment, along with factors such as costs, schedules, environmental consequences, and technical risk will factor into the Record of Decision for the treatment and management of sodium-bonded spent nuclear fuel.

24-5: The high-level radioactive waste form resulting from PUREX process is borosilicate glass, which has already been extensively tested and analyzed under conditions relevant to a geologic repository. The ceramic and metallic

Commentor No. 24: Monte Wilson

Response to Commentor No. 24 (Cont'd):

spent nuclear fuel. The production of a chemically stable waste form to replace a chemically reactive waste form (i.e., sodium-bonded spent nuclear fuel) represents an improvement in the safe, long-term storage of this spent nuclear fuel. DOE expects the new waste forms resulting from the electrometallurgical treatment process will be suitable for disposal in a repository and will meet the requirements of the final waste acceptance criteria.

24-6: The commentor's recommendation of a nonseparation technology is noted. Also noted is the commentor's recommendation for packaging cleaned blanket sodium-bonded spent nuclear fuel in high-integrity cans. At the present time the complete removal of metallic sodium from driver sodium-bonded spent nuclear fuel is not feasible. However, the commentor's recommendation for further development leading to the removal of sodium from driver spent nuclear fuel is noted.

24-7: The commentor's preference for a nonseparation technology to treat sodium-bonded spent nuclear fuel is noted. In addition to the GMODS and direct plasma arc-vitreous ceramic treatment processes, which are considered and dismissed from evaluation in this EIS as less mature technologies, the melt and dilute treatment process is another nonseparation technology. The melt and dilute treatment process is analyzed in this EIS and is being considered for treating driver and blanket fuel at ANL-W and blanket fuel at SRS.

Commentor No. 25: Nancy Fenn

Date: 8/30/99 2:03 PM
Priority: Normal
Subject: Fwd[2]:STOP THE MADNESS!

I am commenting on the Draft Environmental Imapact Statement for The Treatment and Management of Sodium-Bonded Spent Nuclear Fuel.

Please stop nuclear weapons work in Idaho! The proposed project at the INEEL creates new forms of waste, creates nuclear bomb ingredients which runs counter to US nonproliferation goals, and takes valuable money away from greater environmental problems at INEEL. Not to mention that it is a waste of taxpayer dollars. 25-1 25-2 25-3 25-4

Please grant an extension of the comment period of at least 60 days. This extension should be granted because the DOE is missing the following: 25-5

a) the demonstration project results on pyroprocessing 25-6
b) National Academy of Sciences review of the proposed treatment 25-7
c) cost analysis of the various alternatives 25-8
d) nuclear weapons proliferation assessment by the Department of Energy 25-9
e) Yucca Mountain Environmental Impact Statement-the waste acceptance criteria are not known. 25-10

Please, come to your senses before this century ends and do the right thing for Idaho and this country. If nothing else, grant the extension so that you have all relevant information before you make a decision. This country is too great to destroy. 25-11 25-5 25-11

Sincerely,
Nancy W. Fenn

Response to Commentor No. 25:

25-1: As stated in Section 1.1 of the EIS, the proposed action of this EIS is to treat and manage sodium-bonded spent nuclear fuel and facilitate its ultimate disposal in a geologic repository, not to perform nuclear weapons work in Idaho.

25-2: All of the alternatives evaluated in this EIS would produce some forms of high-level radioactive waste. Electrometallurgical treatment (or pyroprocessing) would produce two new waste forms, both of which are more stable than untreated sodium-bonded spent nuclear fuel. DOE expects that these waste forms would be suitable for disposal in a geologic repository.

25-3: The assessment of nonproliferation impacts is not a part of the EIS process. None of the alternatives analyzed in this EIS would generate weapons-usable fissile materials at INEEL. Although highly enriched uranium would be an interim product, it would be downblended to low-enriched uranium during electrometallurgical treatment.

25-4: Congress determines how funds are allocated. DOE spends monies consistent with Congressional direction. DOE is not in a position to make the difficult tradeoffs that may be required between alternative Federal programs and spending priorities. The issue of funding for the treatment and management of sodium-bonded spent nuclear fuel is beyond the scope of the SBSNF EIS.

25-5: In an effort to ensure that all interested parties had time to comment on the draft EIS, the deadline for transmittal of comments was extended from September 13 to September 28, 1999 (64 FR 49169).

25-6: The Electrometallurgical Treatment Research and Demonstration Project began in June 1996 and, although the test results have not been finalized in a single report, a number of status reports issued by the National Academy of Sciences' National Research Council Committee were considered in the preparation of the draft EIS. Success criteria established at the outset of the project have been fulfilled. The environmental impact analysis associated with the electrometallurgical process alternatives was based on actual data from the project. Section 1.6.3 of the EIS summarizes the status and the results of the project.

25-7: The National Academy of Sciences' National Research Council prepared interim status reports on the results of the Electrometallurgical Treatment Research and Demonstration Project that have been reviewed by DOE

Commentor No. 25: Nancy Fenn

Response to Commentor No. 25 (Cont'd):

electrometallurgical treatment demonstration project in September 1999 and published their final summary report in April 2000. The final report findings will be considered during the decision-making process leading to the Record of Decision.

25-8: The SBSNF EIS was prepared in accordance with NEPA, Council on Environmental Quality regulations on implementing NEPA (40 CFR 1500-1508), and DOE's NEPA implementation procedures (10 CFR 1021). None of these require the inclusion of a cost analysis in an EIS. As discussed in the introduction, the basic objective of this EIS is to provide the public and DOE decision-makers with a description of the reasonable alternatives for the treatment and management of DOE's sodium-bonded spent nuclear fuel and their potential environmental impacts. However, DOE issued a separate Cost Study on August 12, 1999, that analyzes and compares the costs of the alternatives analyzed in the EIS. Cost will be considered during the decision-making process in preparing the Record of Decision.

25-9: Although the assessment of nonproliferation impacts is not a part of the scope of the EIS, DOE's Nonproliferation Impacts Assessment was mailed out to interested members of the public on August 12, 1999, and is available by request. The assessment was also placed in the DOE public reading rooms and distributed at the public hearings held during the public comment period on the draft EIS. Information from the assessment, along with other factors such as costs, schedules, environmental consequences, and technical risk will be considered during the decision-making process leading to the Record of Decision.

25-10: No final waste acceptance criteria for a geologic repository have been established at this time. DOE expects that the waste forms described in this EIS will be acceptable. The Draft Yucca Mountain EIS was issued in July 1999 and is discussed in Section 1.6.2.2 of this EIS.

25-11: The scope of this EIS encompasses a comprehensive evaluation of the environmental impacts of alternatives for the treatment and management of sodium-bonded spent nuclear fuel. This EIS indicates that the environmental impacts of using any of the alternatives to treat and manage sodium-bonded spent nuclear fuel are very small. The removal of chemically reactive sodium creates a safer product for disposal in a repository, thus reducing risks to the environment.

Commentor No. 26: John Tanner

Testimony August 26, 1999

The treatment of the driver portion of the sodium-bonded nuclear fuel by the electrometallurgical process is the most sensible option proposed, for the following reasons.

It would allow recovery and use of the high-enriched uranium, which is valuable material that was costly to produce.

This fuel is not suitable for the PUREX process at Savannah River Laboratory because the sodium cannot be completely removed from this fuel by any reasonable process.

26-1

The other methods--melt and dilute, chloride volatility, plasma arc ceramic process, and the glass material process (GMODS)--are less well developed, are likely to be more expensive even after development, and involve heating the fuel to high temperatures, which will worry some people about whether the volatile elements would pollute the air.

The plutonium in the blanket fuel is valuable and should be recovered.

If this were done by the PUREX process, the recovered plutonium would be pure enough to be made into mixed oxide fuel to generate electricity in commercial power reactors. Much of the development of this process is already contemplated for plutonium recovered from weapons. The costs of decladding, sodium removal, and shipment from Idaho would need to be considered.

26-2

The plutonium could also be recovered by the electrometallurgical process. Why is this not mentioned as an alternative in the DEIS? This is as reasonable as many of the other alternatives presented. Although the recovered plutonium would be too contaminated with other transuranic elements to be useful as MOX fuel, it would be useful in a future fast neutron reactor, such as the one which produced it.

But to answer the question just raised, recovery of plutonium by the electrometallurgical process was omitted in order to please influential anti-nuclear critics, who raise weapons proliferation concerns, ignoring the fact that the electrometallurgical process is far more proliferation resistant than the well known PUREX process. The demonstration of plutonium separation by the electrometallurgical process would do nothing to aid anyone's ability to obtain weapons usable material.

26-3

However, putting this plutonium in the waste, as proposed for most of the alternatives in the DEIS, will only temporarily please these critics. When it is later proposed to bury this waste, whether in Yucca Mountain or elsewhere, they will again object, pointing to plutonium's long half-life, and to recent evidence that trace amounts of plutonium can migrate in ground water under special, artificial conditions. Note that the critics have been vehemently opposing the transport and burial of waste with oply trace amounts of plutonium in the WIPP. What will they say when it is proposed to bury waste with substantial amounts of plutonium?

Any method of dealing with plutonium will be criticized, therefore we should do the sensible thing and recover it for later use. J. T.

26-2

Response to Commentor No. 26:

26-1: The commentor's support for electrometallurgical treatment of driver sodium-bonded spent nuclear is noted. The EIS discusses all of the commentor's concerns. Separate studies consider the nonproliferation characteristics of the various alternative technologies and the costs associated with each of the alternatives. The EIS assessment and the conclusions presented in the separate studies will provide some of the information that will be considered during DOE's decision-making process, the results of which will be published in the Record of Decision.

26-2: The commentor's remarks about the value of plutonium present in the sodium-bonded spent nuclear fuel are noted. The intent of this EIS, as discussed in Section 1.2, is to resolve issues associated with the sodium content of sodium-bonded spent nuclear fuel. The disposition of the fissile material content of the fuel is not within the scope of the EIS and is not considered an issue in the formulation of the reasonable alternatives. It is, however, an important consideration in the Nonproliferation Impacts Assessment of the alternatives that was prepared separately from the EIS. The conclusions of the Nonproliferation Impacts Assessment, along with those of the EIS, will be considered during the decision-making process leading to the Record of Decision.

26-3: DOE, consistent with U.S. nuclear nonproliferation policy, would not separate plutonium except for the PUREX process. DOE expects that the plutonium-containing waste from the electrometallurgical treatment process would be acceptable in a geologic repository for the same reasons that plutonium-containing commercial spent nuclear fuel is already acceptable.

Commentor No. 27: John Commander

ELECTROMETALLURGICAL EIS TALKING POINTS FOR AUGUST 26 1999

O I support the treatment of sodium-bonded spent nuclear fuel by the electrometallurgical process. The process should be used for all such fuel, as described in alternative 1 of the Draft Environmental Impact Statement. **27-1**

O The electrometallurgical treatment has been proven to be satisfactory. Many of the other alternatives are in the concept or research stage.

O Nearly all the sodium-bonded fuel is now at ANL-West. It makes both common and economic sense to do the entire treatment there. **27-2**

O I am concerned about the loss of jobs and skills, if the treatment is not at ANL-W. These skills are particularly important at this time. The current administration is finally putting some new funding into nuclear research and technology. DOE has designated INEEL as the lead laboratory for this effort. **27-3**

O The electrometallurgical treatment has little risk that nuclear material could be diverted to use in nuclear bombs. The Draft EIS has adequately answered the comments of those concerned about that risk. **27-4**

O Whatever alternative is chosen, it must meet the terms of the 1995 Governor's Agreement on Nuclear Waste. If treatment is done at Savannah River, material must be moved there before the Year 2035. This date is the deadline for all spent fuel to be out of Idaho. **27-5**

Submitted by:

John Commander.
Member: IANS & INEL Retired Employees Association
Telephone (208) 523 5788

Response to Commentor No. 27:

27-1: The commentor's support for the electrometallurgical treatment of both driver and blanket sodium-bonded spent nuclear fuel (Alternative 1) is noted.

27-2: The commentor's support for treatment of all sodium-bonded spent nuclear fuel at ANL-W is noted. The cost implications compared to other alternatives are evaluated in a separate Cost Study.

27-3: The commentor's concern about the loss of jobs and skills if treatment of sodium-bonded spent nuclear fuel is not conducted at ANL-W is noted. DOE recognizes the value and the presence of important skills at ANL-W and INEEL. As part of the decision-making process, DOE will consider the consequences of potential impacts to various environmental resources, including socioeconomics. The Record of Decision will explain the rationale and factors for DOE's decision.

27-4: The commentor is correct. Under this specific application, electrometallurgical treatment of sodium-bonded spent nuclear fuel would not produce weapons-usable material, thereby reducing the risk that this spent nuclear fuel might be diverted for other uses.

27-5: The terms of the State of Idaho Settlement Agreement and Consent Order (Governor's Agreement) are accounted for in all of the alternatives evaluated in this EIS. A copy of the agreement is provided in Appendix K.

Commentor No. 28: Terry & Theresa Williams

Terry & Theresa Williams
P. O. Box 1627
Hailey, ID 83333

August 25, 1999

Ms. Susan Lesica
U.S. Department of Energy
Office of Nuclear Facilities Management, NE-40
19901 Germantown Road
Germantown, Maryland 20874-1290

Dear Ms. Lesica:

We are writing to register our comments on the "Draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel."

We request that you extend the comment period past September 13, 1999 by at least 60 days since much of the information relevant to the procedure won't be available until after the comment period is closed. 28-1

In closing, we would like to add that we are against Pyroprocessing. 28-2

Thank you very much for registering our comments.

Terry & Theresa Williams

Response to Commentor No. 28:

28-1: In an effort to ensure that all interested parties had time to comment on the draft EIS, the deadline for transmittal of comments was extended from September 13 to September 28, 1999 (64 FR 49169). DOE made materials relevant to the review of the draft EIS available in public reading rooms and at a series of public hearings that were held on August 17, 1999, in North Augusta, South Carolina; August 24, 1999, in Boise, Idaho; August 26, 1999, in Idaho Falls, Idaho; and August 31, 1999, in Arlington, Virginia. The materials placed in the reading rooms included the electrometallurgical demonstration project environmental assessment, the Finding of No Significant Impact for the environmental assessment, National Research Council interim status reports on the demonstration project, the 1995 Settlement Agreement and Consent Order with the State of Idaho, the EIS scoping meeting transcripts and public hearing comments, and the draft EIS hearing presentations and fact sheets. In addition, completion of the Cost Study and Nonproliferation Impacts Assessment was expedited so that they would be available to the public during the comment period. These reports were mailed to interested parties on August 12, 1999, and were made available to the public at the public hearings on the draft EIS. Although these reports are not required for the environmental impact analysis presented in the EIS, they will provide input to the Record of Decision. While the final National Research Council report on the electrometallurgical treatment demonstration project at ANL-W was published in April 2000, interim status reports were produced throughout the project and this data was used to prepare the EIS.

28-2: The commentor's objection to electrometallurgical treatment (pyroprocessing) of sodium-bonded spent nuclear fuel is noted.

Commentor No. 29: Robert H. Wilcox

_____Forward Header_____
Subject: DOE/EIS-0306D, Draft EIS for Treatment & Management of Na-Bonded Spent Nuclear Fuel
Author: RBTHWILCOX@aol.com_at_INTERNET
Date: 9/6/99 3:30 PM

William D Magwood, IV, Director
Office of Nuclear Energy, Science & Technology
U. S. Department of Energry (DOE)

In my opinion, the DOE has outdone itself in following blindly, once again, its ill-advised process of using EIS's instead of sound management to carry out its business. **29-1**

In this instance, it has even "no preferred alternative at this time". And, it has determined that the alternatives evaluated in detail would have minimal or no impacts n the remaining environmental areas. **29-2**
Really! Must we continue to waste the taxpayers money on such analyses as opposed to getting on with the job at hand? **29-3**

As an aside, it is intriguing to note that DOE appears to feel committed to its 1995 agreement with the State of Idaho to remove all SNF from Idaho by 2035. Yet, DOE has appeared singularly uninterested in honoring its legal commitment to accept custody of SNF from the nation's power reactors, a far more urgent and pressing matter. **29-4**

My recommendations are:
1. With respect to Sodium-Bonded SNF: Get on with doing this in Idaho, which was the plan from day one. If given the necessary attention, DOE surely ought to be able to accomplish this by 2035. (How long did it take us to put a man on the moon?) **29-5**

2. Please, please reexamine the need to WASTE significant portions of the DOE budget on the stupid EIS process. I'm not saying that the environment should be ignored, only that DOE needs to be more action-oriented. Sure this might need new legislation, but when was the last time anyone in DOE suggested to the CEQ that that might be a good idea? **29-3**

If anyone there would like further suggestions on this subject, or clarification of the above, from this reviewer, please let me know.

Yours truly,

Robert H. Wilcox,
Former USAEC Employee, Retired Foreign Service Officer & SRS Employee,
711 Pevero Abbey Circle, Martinez, GA 30907
Tel.: (706) 855-5824

Response to Commentor No. 29 :

29-1: DOE is required under NEPA to prepare an EIS when its actions could significantly affect the environment, as in the case of the treatment and management of DOE's sodium-bonded spent nuclear fuel. In its Finding of No Significant Impact for the environmental assessment of the Electrometallurgical Treatment Research and Demonstration Project (May 1996), DOE committed to preparing an EIS before making any significant additional use of the electrometallurgical treatment technology. DOE strongly believes that preparation of this EIS is consistent with sound management principles and its policy of fully informing both decision-makers and the public of the potential environmental consequences of any proposed action.

29-2: Council on Environmental Quality regulations (40 CFR 1502.14[e]) do not require a preferred alternative to be included in a draft EIS if one has not been identified at the time of publication. However, the regulations do require that a preferred alternative be identified in a final EIS. Section 2.8 of this EIS identifies the Preferred Alternative. The reader's comment related to minimal or no impacts is noted.

29-3: Congress determines how funds are allocated. DOE spends monies consistent with Congressional direction. DOE is not in the position to make the difficult tradeoffs that may be required between alternative Federal programs and spending priorities. The issue of spending money for the treatment and management of sodium-bonded spent nuclear fuel is beyond the scope of the SBSNF EIS.

29-4: The scope of this EIS is for the treatment and management of sodium-bonded spent nuclear fuel only. It does not include commercial nuclear power spent nuclear fuel. However, it should be noted that some of the sodium-bonded spent nuclear fuel was generated by the Fermi-1 commercial power reactor, which operated in the 1960s. In addition, DOE has issued a draft EIS for the Yucca Mountain waste repository which does address the disposal of commercial spent nuclear fuel.

29-5: The commentor's support for treating the sodium-bonded spent nuclear fuel at INEEL is noted.

Commentor No. 30: David Kipping

_____Forward Header_____
Subject: Comment on Pyroprocessing EIS
Author: kipping@compuserve.com_at_INTERNET
Date: 9/7/99 5:49 PM

Message authorized by: kipping@compuserve.com_at_INTERNET at X400PO

Dear Ms. Lesica,

Ref: "Draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel"

It is essential that the comment period for the EIS be extended a minimum of 60 days (to at least 15 Nov 99). This draft EIS is very premature; much of the important information the public needs to make informed comments will not be ready for at least another month. Therefore the September 13 deadline does not allow enough time. **30-1** **30-2**

There is a great deal of information that is essential to making sound technical and policy decisions that is not in the draft EIS and, in fact, is not available at all. In particular, the following items are missing:

- The demonstration project at Argonne West on the proposed treatment--final results will not be available until the end of the September at the earliest. **30-3**
- National Academy of Sciences review of the proposed treatment. **30-4**
- Cost analysis of the various alternatives. **30-5**
- Nuclear weapons proliferation assessment by the Department of Energy . It is my understanding that this was just released last week. **30-6**
- Yucca Mountain (the purported destination of pyroprocessing waste) Environmental Impact Statement--the waste acceptance criteria are not known. **30-7**

The current draft EIS is seriously flawed due to the lack of information mentioned above. As a minimum, DOE needs to wait until all this information is available before closing the comment period. But merely extending the comment period will not achieve the result desired: an EIS that presents all the facts and allows the public to make informed comments. Much of the public will not be aware of the additional documents and/or will not have access to them. It is clear that the best approach is incorporate the above-mentioned information in a second draft EIS which then can be made available for well-informed public comment. **30-2** **30-8**

Thank you for your consideration.

David Kipping
Pres[illegible]dent, Board of Directors

Response to Commentor No. 30:

30-1: In an effort to ensure that all interested parties had time to comment on the draft EIS, the deadline for transmittal of comments was extended from September 13 to September 28, 1999 (64 FR 49169).

30-2: DOE does not believe that the draft EIS was produced prematurely because of a failure to present all the facts necessary for the public to make informed comments. However, DOE did extend the comment period to ensure that all interested parties had time to adequately review the draft document (64 FR 4916). DOE made material supporting the preparation of the EIS available in public reading rooms and through a series of public hearings held August 17, 1999, in North Augusta, South Carolina; August 24, 1999, in Boise, Idaho; August 26, 1999, in Idaho Falls, Idaho; and August 31, 1999, in Arlington, Virginia. Materials placed in the public reading rooms included the environmental assessment for the Electrometallurgical Treatment Research and Demonstration Project, the Finding of No Significant Impact for the environmental assessment, National Research Council reports, the 1995 Settlement Agreement and Consent Order with the State of Idaho, scoping period meeting transcripts and comments, and the draft EIS hearing presentations and fact sheets. In addition, completion of the Cost Study and Nonproliferation Impacts Assessment was expedited so that these also would be available to the public during the comment period. These reports were mailed to interested parties on August 12, 1999, and were made available to the public at the public hearings on the draft EIS. Although these reports are not critical to the environmental impact analysis presented in the EIS, they will be considered during the decision-making process leading to the Record of Decision. While the final National Research Council report on the Electrometallurgical Treatment Research and Demonstration Project at ANL-W was published in April 2000, interim status reports were produced throughout the project. Data generated during the demonstration project was used in preparing the EIS, as discussed in Section 1.6.3 of the EIS.

30-3: Final test results were made available in August 1999 and were used in the EIS. The success criteria established at the outset of the project were fulfilled. The environmental impact analysis associated with the electrometallurgical process alternatives was based on actual data from the project. Section 1.6.3 of the EIS summarizes the status and the results of the project.

Commentor No. 30: David Kipping

Response to Commentor No. 30: (Cont'd)

Research and Demonstration Project that have been reviewed by DOE. The National Research Council completed their evaluation of the electrometallurgical treatment demonstration project in September 1999 and published their final summary report in April 2000. The final report findings will be considered during the decision-making process leading to the Record of Decision.

30-5: The SBSNF EIS was prepared in accordance with NEPA, Council on Environmental Quality regulations on implementing NEPA (40 CFR 1500-1508), and DOE's NEPA implementation procedures (10 CFR 1021). None of these require the inclusion of a cost analysis in an EIS. As discussed in the introduction, the basic objective of this EIS is to provide the public and DOE decision-makers with a description of the reasonable alternatives for the treatment and management of DOE's sodium-bonded spent nuclear fuel and their potential environmental impacts. However, DOE issued a separate Cost Study on August 12, 1999, that analyzes and compares the costs of the alternatives analyzed in the EIS. Cost will be considered during the decision-making process in preparing the Record of Decision.

30-6: Although the assessment of nonproliferation impacts is not a part of the EIS process, DOE's Nonproliferation Impacts Assessment was mailed out to interested members of the public on August 12, 1999, and is available by request. The assessment was also placed in the DOE public reading rooms and distributed at the public hearings held during the public comment period on the draft EIS. Information from the assessment, along with other factors such as costs, schedules, environmental consequences, and technical risk will be considered during the decision-making process leading to the Record of Decision.

30-7: As discussed in Section 2.7 of this EIS, final waste acceptance criteria for a geologic repository are still being developed. DOE expects the waste forms that would be produced by the proposed action would be suitable for disposal in a geologic repository. In July 1999, DOE published a Draft Yucca Mountain EIS, which is discussed in Section 1.6.2.2 of this EIS. The Yucca Mountain EIS assumes that sodium-bonded spent nuclear fuel is treated using the electrometallurgical process prior to emplacement in the geologic repository.

30-8: DOE has made material supporting the preparation of the EIS available in public reading rooms and through a series of public hearings which were advertised in the Federal Register, as well as local newspapers. In addition, completion of the Cost Study and Nonproliferation Impacts Assessment was expedited so that these would be available to the public during the

Commentor No. 30: David Kipping

Response to Commentor No. 30 (Cont'd):

comment period. These reports were mailed to interested parties on August 12, 1999, and were made available to attendees at all of the public hearings on the draft EIS. While the final National Research Council report on the Electrometallurgical Treatment Research and Demonstration Project at ANL-W was published in April 2000, interim status reports have been produced throughout the project and these are available in the public reading rooms. Considering the additional time provided by the extension of the comment period and the availability of the data used to prepare the EIS, DOE does not feel that a second draft is warranted.

Commentor No. 31: David Hensel

09/06/1999 21:41 2083548636 HENSEL/ROBINSON PAGE 1

David Hensel
PO Box 1104
313 S. 200E.
Driggs, Id.
83422
208-354-8636
hensel@nationvalley.net

Dear Ms. Lesica

I want to comment on the DEIS for treatment and management of sodium-bonded spent nuclear fuel.

First, I wish to ask for an extension. Far too much information is missing from the deis: 31-1

1. the demonstration results on pyroprocessing 31-2
2. NAS's review of the proposed treatment 31-3
3. Cost analysis of the alternatives 31-4
4. Nuclear weapons proliferation's risk assessment 31-5
5. The acceptance criteria for Yucca Mt., and the EIS on Yucca Mt. 31-6

Pyroprocessing is a bad idea. It separates out nuclear bomb grade uranium from spent fuel. Developing this technology runs counter to the USA's nonproliferation goals. The technology can be modified to separate out bomb-grade plutonium. There is nothing more fleeting than a military secret. Witness the uproar over China's stealing our bomb making secrets. Why spend tax dollars developing technology that eventually will find its way into unfriendly hands. Further developing this technology sends the rest of the world the message that the US is not serious about stopping the spread of nuclear weapons technology. 31-7

As the DOE is aware, I hope, the INEEL is awash in extremely dangerous waste. Waste that it lacks the technology and resources to clean up. Much of this waste was produced during reprocessing. Rather than dealing with the crucial problem of cleanup, the DOE now proposes to produce more waste? This makes no sense. The cleanup budget is flat for the foreseeable future; many projects have no funding (the liquid waste tanks at the INEEL for instance). Pyroprocessing takes money that could be used much better on environmental projects. There is plenty, too much, weapons grade material and nuclear waste already-this project only produces more of the same. It is a waste of precious resources and taxpayer money. 31-8 31-9 31-7 31-8 31-9

Thank you,

David Hensel

Response to Commentor No. 31:

31-1: In an effort to ensure that all interested parties had time to comment on the draft EIS, the deadline for transmittal of comments was extended from September 13 to September 28, 1999 (64 FR 49169).

31-2: The Electrometallurgical Treatment Research and Demonstration Project began in June 1996 and, although the test results have not been finalized in a single report, a number of status reports issued by the National Academy of Sciences' National Research Council Committee were considered in the preparation of the draft EIS. The success criteria established at the outset of the project were fulfilled. The environmental impact analysis associated with the electrometallurgical process alternatives was based on actual data from the project. Section 1.6.3 of the EIS summarizes the status and the results of the project.

31-3: The National Academy of Sciences' National Research Council prepared interim status reports on the results of the Electrometallurgical Treatment Research and Demonstration Project that have been reviewed by DOE. The National Research Council completed their evaluation of the electrometallurgical treatment demonstration project in September 1999 and published their final summary report in April 2000. The final report findings will be considered during the decision-making process leading to the Record of Decision.

31-4: The SBSNF EIS was prepared in accordance with NEPA, Council on Environmental Quality regulations on implementing NEPA (40 CFR 1500-1508), and DOE's NEPA implementation procedures (10 CFR 1021). None of these require the inclusion of a cost analysis in an EIS. As discussed in the introduction, the basic objective of this EIS is to provide the public and DOE decision-makers with a description of the reasonable alternatives for the treatment and management of DOE's sodium-bonded spent nuclear fuel and their potential environmental impacts. However, DOE issued a separate Cost Study on August 12, 1999, that analyzes and compares the costs of the alternatives analyzed in the EIS. Cost will be considered during the decision-making process in preparing the Record of Decision.

31-5: Although the assessment of nonproliferation impacts is not a part of the EIS process, DOE's Nonproliferation Impacts Assessment was mailed out to interested members of the public on August 12, 1999, and is available by request. The assessment was also placed in the DOE public reading rooms and distributed at the public hearings held during the public comment period on the draft EIS. Information from the assessment, along with other factors such as costs, schedules, environmental

Commentor No. 31: David Hensel

Response to Commentor No. 31 (Cont'd):

consequences, and technical risk will be considered during the decision-making process leading to the Record of Decision.

31-6: As discussed in Section 2.7 of this EIS, final waste acceptance criteria are still being developed for a geologic repository. DOE expects the waste forms produced by the proposed action would be suitable for disposal in a geologic repository. In July 1999, DOE published a Draft Yucca Mountain EIS, which is discussed in Section 1.6.2.2 of this EIS. The Yucca Mountain EIS assumes that sodium-bonded spent nuclear fuel is treated using the electrometallurgical process prior to emplacement in the geologic repository.

31-7: The assessment of nonproliferation impacts is not a part of the EIS process. None of the alternatives analyzed in this EIS would generate weapons-usable fissile materials at INEEL. Although highly enriched uranium would be an interim product, it is would be down-blended to low-enriched uranium during electrometallurgical treatment.

31-8: The sodium-bonded spent nuclear fuel at INEEL contains metallic sodium, which is chemically reactive and so can be a potentially dangerous substance in the spent nuclear fuel. This EIS evaluates the impacts of treating and managing this sodium-bonded spent nuclear fuel so that, for the analyzed alternatives, this chemically reactive and potentially dangerous sodium is removed or converted to a nonreactive form. Such treatment would reduce the danger of radioactive material releases to the environment from emplacement of this radioactive material in a geologic repository. The environmental impact of waste generated from the proposed action is addressed in Chapter 4 of the EIS.

31-9: Congress determines how funds are allocated. DOE spends monies consistent with Congressional direction. DOE is not in a position to make the difficult tradeoffs that may be required between alternative Federal programs and spending priorities. The issue of funding for the treatment and management of sodium-bonded spent nuclear fuel is beyond the scope of the SBSNF EIS.

Commentor No. 32: Lowell Jobe

SB SNF Toll Free Line

9/8/99

Lowell Jobe
Coalition 21
14469 N 55th East
Idaho Falls, ID 83401
hm: 208-524-7271
fax: 208-524-0998

Coalition 21: 208-542-1575

I am calling to ask if there will be an extension on the comment period. We have found some serious questions regarding the cost report figures and tables and feel they need answering before we can finalize our conclusions and comments. Please advise us by phone. Please leave an answer on our answering machines if we are not there. 32-1 32-2

Response to Commentor No. 32:

32-1: In an effort to ensure that all interested parties had time to comment on the Draft EIS, the deadline for transmittal of comments was extended from September 13, 1999, to September 28, 1999 (64 FR 49169).

32-2: Actual costs for treating and managing sodium-bonded spent nuclear fuel are not part of the scope of the EIS. DOE welcomes questions concerning the August 1999 Cost Study.

Commentor No. 33: Lisa Johnson

Ms. Susan Lesica
US Department of Energy
Office of Nuclear Facilities Management, NE-40
19901 Germantown Road
Germantown, Maryland 20874-1290

Dear Ms. Lesica,

I am commenting on the DEIS for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel.

Our family, which includes young children, lives downwind of the Idaho National Engineering and Environmental Lab. I am very concerned about our family's health and the quality of the environment in the region due to the many activities that occur at the lab. 33-1

I have several comments about this project and DEIS. The general feeling among our friends is that pyroprocessing is not an acceptable project for INEEL because of the following factors: 33-2

- It creates new forms of nuclear waste, an issue that is already a huge problem at INEEL. 33-3
- It takes money away from greater environmental problems at INEEL. 33-4
- It wastes taxpayer money (Don't forget it has been mentioned twice on NBC new's "Fleecing of America"). 33-5
- And it creates bomb-grade uranium from spent fuel and thus runs counter to US nonproliferation goals.

It seems clear that a 60-day extension of the comment period is necessary because so much relevant information is not yet available. For example: 33-6

- The DOE does not have the results of the demonstration project on pyroprocessing. 33-7
- The review from the National Academy of Sciences of the proposed treatment is not yet available. 33-8
- The Department of Energy's nuclear weapons proliferation assessment is not included. 33-9
- The waste acceptance criteria is not known because the Yucca Mountain EIS is not included (the purported destination of the waste). 33-10

Once again, I do not support pyroprocessing at INEEL and a 60-day extension of the comment period is necessary because so much pertinent information is missing from the document. 33-2, 33-6

Sincerely,

Lisa Johnson
PO Box 542
Victor ID 83455

Response to Commentor No. 33:

33-1: As indicated in the EIS, the human health effects resulting from operational activities to treat and manage the sodium-bonded fuel are very small. The estimated cumulative health effects to the public residing in the vicinity of INEEL from current and reasonably foreseeable future activities are summarized in Section 4.11.1.4 of the EIS. As indicated in this section, the expected health effects from these activities are very small. For example, an individual residing at the INEEL site boundary would be expected to receive a maximum radiation dose of 0.4 millirem per year from all releases, compared to natural background doses of 360 millirem per year, and are well below the regulatory limit of 10 millirem per year. Appendix E, Section E.2.1, of the EIS provides the Federal and DOE regulatory limits on radiation exposures.

33-2: The commentor's objection to electrometallurgical treatment (pyroprocessing) of sodium-bonded spent nuclear fuel at INEEL is noted.

33-3: All of the alternatives evaluated in this EIS would produce some forms of high-level radioactive waste. Electrometallurgical treatment (or pyroprocessing) would produce two new waste forms, both of which are more stable than untreated sodium-bonded spent nuclear fuel. DOE expects that these waste forms would be suitable for disposal in a geologic repository. Treatment of current high-level radioactive waste at INEEL is being evaluated in the Idaho High-Level Waste and Facilities Disposition Draft EIS, which is discussed in Section 1.6.2.3 of this EIS.

33-4: Congress determines how funds are allocated. DOE spends monies consistent with Congressional direction. DOE is not in a position to make the difficult tradeoffs that may be required between alternative Federal programs and spending priorities. The issue of funding for the treatment and management of sodium-bonded spent nuclear fuel is beyond the scope of the SBSNF EIS.

33-5: The assessment of nonproliferation impacts is not a part of the scope of the EIS. None of the alternatives analyzed in this EIS would generate weapons-usable fissile materials at INEEL. Although highly enriched uranium would be an interim product, it would be down-blended to low-enriched uranium during electrometallurgical treatment.

33-6: In an effort to ensure that all interested parties had time to comment on the draft EIS, the deadline for transmittal of comments was extended from September 13 to September 28, 1999 (64 FR 49169). DOE made materials

Commentor No. 33: Lisa Johnson

Response to Commentor No. 33 (Cont'd):

Augusta, South Carolina; August 24, 1999, in Boise, Idaho; August 26, 1999, in Idaho Falls, Idaho; and August 31, 1999, in Arlington, Virginia. The materials placed in the reading rooms included the electrometallurgical demonstration project environmental assessment, the Finding of No Significant Impact for the environmental assessment, National Research Council interim status reports on the demonstration project, the 1995 Settlement Agreement and Consent Order with the State of Idaho, the EIS scoping meeting transcripts and public hearing comments, and the draft EIS hearing presentations and fact sheets. In addition, completion of the Cost Study and Nonproliferation Impacts Assessment was expedited so that they would be available to the public during the comment period. These reports were mailed to interested parties on August 12, 1999, and were made available to the public at the public hearings on the draft EIS. Although these reports are not critical to the environmental impact analysis presented in the EIS, they will provide input to the Record of Decision. While the final National Research Council report on the electrometallurgical treatment demonstration project at ANL-W was published in April 2000, interim status reports were produced throughout the project and this data was used to prepare the EIS as discussed in Section 1.6.3 of the EIS.

33-7: Final test results were made available in August 1999 and were used in the EIS. The success criteria established at the outset of the project were fulfilled. The environmental impact analysis associated with the electrometallurgical process alternatives was based on actual data from the project. Section 1.6.3 of the EIS summarizes the status and the results of the project.

33-8: The National Academy of Sciences' National Research Council prepared interim status reports on the results of the Electrometallurgical Treatment Research and Demonstration Project that have been reviewed by DOE. The National Research Council completed their evaluation of the electrometallurgical treatment demonstration project in September 1999 and published their final summary report in April 2000. The final report findings will be considered during the decision-making process leading to the Record of Decision.

33-9: Although the assessment of nonproliferation impacts is not a part of the EIS process, DOE's Nonproliferation Impacts Assessment was mailed out to interested members of the public on August 12, 1999, and is available by request. The assessment was also placed in the DOE public reading rooms and distributed at the public hearings held during the public comment period

Commentor No. 33: Lisa Johnson

Response to Commentor No. 33 (Cont'd):

on the draft EIS. Information from the assessment, along with other factors such as costs, schedules, environmental consequences, and technical risk will be considered during the decision-making process leading to the Record of Decision.

33-10: As discussed in Section 2.7 of the EIS, final waste acceptance criteria for a geologic repository are still being developed. DOE expects the waste forms that would be produced by the proposed action would be suitable for disposal in a geologic repository. In July 1999, DOE published a Draft Yucca Mountain EIS, which is discussed in Section 1.6.2.2 of this EIS. The Yucca Mountain EIS assumes that sodium-bonded spent nuclear fuel is treated using the electrometallurgical process prior to emplacement in the geologic repository.

Commentor No. 34: Dan Johnston

From: Daniel.C.Johnston@rl.Doe.gov_at_INTERNET at X400PO
Date: 9/13/99 8:37PM -0700
To: EMTEIS at NE-02
*cc: dcjohnston@rl.doe.gov_at_INTERNET at X400PO
Subject: Comments to Na-bonded Fuel EIS

Please see the following comments.

Comments to the Draft Environmental Impact Statement for the
Treatment and Management of Sodium-Bonded Spent Nuclear Fuel

(Doc No DOE EIS-0306D, July 1999)

The document appears complete and addresses the FFTF Sodium-Bonded fuel. The transportation issues are not specifically identified, but the different radiation and risk values shown for the various alternatives indicate they have been addressed. 34-1

My only major concern is what appears to be a difference in how the PU and U are handled as the sodium-bonded spent nuclear fuel is run through the Electro-metallurgical Treatment Process as described in paragraph S.3.1 on page S-13. The PU is lumped in with the other fission products whereas the U is specifically collected and removed, then diluted as necessary. 34-2

Does this uncontrolled gathering of PU guarantee sufficient criticality control for the PU in this process, or should the likelihood of pockets of PU and PU compounds of varying concentrations be acknowledged and monitored with identified actions to be taken to ensure safe handling?

From: Dan Johnston
1471 Amon Ct.
Richland, Wa. 99352

Response to Commentor No. 34:

34-1: As stated in Sections 2.2.3 and 4.2 of this EIS, pursuant to the amended Record of Decision for the DOE Programmatic Spent Nuclear Fuel EIS (61 FR 9441), the sodium-bonded Fast Flux Test Facility fuel would be transported from Hanford to INEEL. The environmental impacts associated with transport of the Fast Flux Test Facility fuel to INEEL are summarized in Appendix G of this EIS by referencing the Programmatic Spent Nuclear Fuel EIS.

34-2: As stated in Appendix C, Section C.1 of the EIS, during electrometallurgical treatment of the sodium-bonded fuel, there are strict criticality controls in place for all aspects of the process. In the electrorefiner, the plutonium would be in a chloride compound in liquid state and would be homogeneously mixed with the other salts. Abnormal localized concentrations of plutonium within the electrorefiner have been analyzed for a number of scenarios. These analyses have confirmed that an adequate margin of criticality safety would exist even under these conditions. Nevertheless, actual operations would carefully monitor the level of plutonium at all stages of the process in order to ensure the early detection of any abnormal conditions that should arise. The concentration of plutonium in the salt would be monitored through repeated sampling. When the salt is stabilized into the ceramic waste, the transuranic and fission products would be uniformly distributed throughout the waste form, which has been confirmed by sampling. The maximum plutonium concentration in the salt would be about 8 weight percent. A conservative criticality assessment was performed on the ceramic waste form. The results of this assessment showed that the plutonium concentration in the waste form would pose no criticality safety concerns.

Commentor No. 35: Carol Murphy

09/13/1999 13:19 208/[illegible]

Carol Murphy
PO Box 4714
Ketchum ID 83340
(208) 726 5124

TO: MS SUSAN LESICA
US DEPT of ENERGY
OFFICE OF NUCLEAR FACILITIES MGMNT.
19901 GERMANTOWN RD.
GERMANTOWN, MD. 20874-1290

Dear Ms. Lesica,

My family and I are opposed to pyroprocessing at the INEEL in Idaho for several reasons. First, it is a tremendous waste of tax payers dollars. Second, we do not need new forms of nuclear waste, which this process will create. Third, the money should be diverted & used elsewhere for more pressing environmental problems at the INEEL. Last, we do not need to be seperating our more 'bomb-grade' uranium.

We also are urging you to extend the comment period for 60 to 90 days or until well after the Yucca Mountain EIS has been released. This is important because if Yucca Mtn. is to be the destination of the pyro-processing waste, we cannot rationally evaluate it until we have an EIS from Yucca Mtn. We also need cost evaluations from all alternatives.

Too much is unanswered, unspecified and unstudied. Too much is not documented and not reviewed.

Please extend the comment period past Sept. 28, 1999 and note that we are opposed to pyro-processing in general. Thank you, Carol Murphy + family

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Response to Commentor No. 35:

35-1: The commentor's objection to electrometallurgical treatment (pyroprocessing) of sodium-bonded spent nuclear fuel at INEEL is noted.

35-2: Congress determines how funds are allocated. DOE spends monies consistent with Congressional direction. DOE is not in a position to make the difficult tradeoffs that may be required between alternative Federal programs and spending priorities. Although not within the scope of this EIS, a separate Cost Study of the alternatives analyzed in the EIS has been developed and is available to the public. This Cost Study evaluates the cost of each alternative, including no action.

35-3: All of the alternatives evaluated in this EIS would produce some forms of high-level radioactive waste. Electrometallurgical treatment (pyroprocessing) would produce two new waste forms, both of which are more stable than untreated sodium-bonded spent nuclear fuel. DOE expects that these waste forms would be suitable for disposal in a geologic repository. Treatment of current high-level radioactive waste at INEEL is being evaluated in the Idaho High-Level Waste and Facilities Disposition EIS, which is discussed in Section 1.6.2 of this EIS.

35-4: None of the alternatives analyzed in this EIS would generate weapons-usable fissile materials at INEEL. Although highly enriched uranium would be an interim product, it would be down-blended to low-enriched uranium during electrometallurgical treatment.

35-5: In an effort to ensure that all interested parties had time to comment on the draft EIS, the deadline for transmittal of comments was extended from September 13 to September 28, 1999 (64 FR 49169).

35-6: The Yucca Mountain Draft EIS was released in July 1999. Relevant information from the Yucca Mountain Draft EIS was incorporated into Section 1.6.2 of this SBSNF EIS.

35-7 DOE issued a separate Cost Study that analyzes and compares the cost of alternatives analyzed in the EIS. Cost will be considered during the decision-making process in preparing the Record of Decision

35-8: The information needed to make a decision concerning the treatment and management of DOE's sodium-bonded spent nuclear fuel was obtained and analyzed in the EIS. This information included input from the public, as well as from Federal, state and local agencies, and Tribal governments. Also included was site-specific information on the environmental conditions

Commentor No. 35: Carol Murphy

Response to Commentor No. 35: (Cont'd)

DOE's Electrometallurgical Treatment Research and Demonstration Project were used to prepare the EIS. The results of this project are documented in a series of reports published by ANL-W and reviewed by the National Research Council. All of the materials used to prepare the EIS are referenced at the end of each chapter.

Commentor No. 36: David E. Adelman

September 13, 1999

Ms. Susan Lesica
U.S. Department of Energy
Office of Nuclear Facilities Management, NE-40
19901 Germantown Road
Germantown, Maryland 20874-1290

Dear Ms. Lesica:

Please find the enclosed comments of the Natural Resources Defense Council, Inc., on the Department of Energy's Draft Environment Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel (DOE/EIS-0306D). Should you have any questions, I can be reached at (202) 289-6868. Thank you very much for your assistance.

Sincerely,

David E. Adelman
Project Attorney, Nuclear Program

Response to Commentor No. 36:

Commentor No. 36: David E. Adelman (Cont'd)

Comments of the Natural Resources Defense Council on The Department of Energy's Draft Environment Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel

The Draft Environmental Impact Statement ("EIS") for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel represents a substantial improvement from the Notice of Intent for the EIS. In particular, the EIS treats separately the analysis of high-burnup driver fuel elements and low-burnup blanket fuel elements. This is a critical distinction to make because the sodium in blanket fuel can be readily removed using mechanical methods. In addition, although not complete, DOE has broadened the alternatives analysis by considering technologies beyond electrometallurgical treatment ("EMT") and PUREX reprocessing. The inclusion of direct disposal of blanket fuel elements and melt-and-dilute treatment of blanket and driver fuel elements are of particular importance because of their reduced environmental and non-proliferation impacts. 36-1

A number of deficiencies persist, both in terms of the process and the substance of the EIS itself. First, the EIS process is proceeding despite the pending National Academy of Sciences report evaluating the EMT technology. The EIS would benefit substantially from the determinations of the Academy's report, particularly in assessing its viability and impacts relative to other methods. DOE has also arbitrarily prepared separate reports on the costs and non-proliferation implications of the alternatives considered in the EIS, and released them without any opportunity for public comment. These assessments should have been incorporated into the Draft EIS and released for public comment. Further research and development of the alternatives considered should also be completed prior to finalizing the EIS to ensure that their viability and environmental impacts are considered fully. 36-2 36-3 36-4

1. DOE Marginalizes Methods Other Than EMT for Processing Driver Fuel Elements

DOE's primary justification for proposing treatment of sodium-bonded fuel using the EMT technology is the reactivity of the bonded-sodium in the fuel, which according to DOE precludes direct disposal of these fuel elements. As DOE has acknowledged, this justification applies, at most, to the driver fuel elements because the sodium can be removed mechanically from the blanket fuel, making it acceptable for direct disposal in a geologic repository. Blanket fuel constitutes more than 95 percent of the fuel to be processed; driver fuel accounts for only three tons of the total 60 tons in storage. With such a limited amount of material to be processed, it makes little sense to invest the time and money in the EMT technology, particularly given the non-proliferation risks associated with its capacity to be used for plutonium extraction.[1] 36-5

[1] The National Academy of Sciences has already expressed concern about this risk: "Although the developers of the electrometallurgical technique argue that the technology is proliferation resistant, any SNF processing approach that is capable of separating fissionable materials from associated fission products and transuranic elements could be redirected to produce material with nuclear detonation capability .... Demonstration of the process could, however, add to the risk that a nation intent on weapons production might consider adapting this technology for possible production of fissile material, although such material would be of poor quality for a weapon." Fred Bazolo et al., *An Assessment of Continued R&D Into an Electrometallurgical Approach for Treating DOE Spent Nuclear Fuel*, S-2 (National Research Council, 1995).

Response to Commentor No. 36 (Cont'd):

36-1: The comment is noted. DOE revised the scope of the EIS based on comments provided during the public scoping period.

36-2: The National Academy of Sciences' National Research Council prepared interim status reports on the results of the Electrometallurgical Treatment Research and Demonstration Project that have been reviewed by DOE. The National Research Council completed their evaluation of the electrometallurgical treatment demonstration project in September 1999 and published their final summary report in April 2000. The final report findings will be considered during the decision-making process leading to the Record of Decision.

36-3: The Cost Study and Nonproliferation Impacts Assessment were prepared to provide additional pertinent information to the Secretary of Energy so that he may make an informed decision concerning the treatment and management of DOE's sodium-bonded spent nuclear fuel. These documents were mailed to interested parties on August 12, 1999, and were made available to attendees at all of the public hearings on the draft EIS. It should be noted that, although NEPA does not require inclusion of the information provided in the Cost Study and the Nonproliferation Impacts Assessment in the EIS, this information will be considered along with other pertinent data when the Record of Decision is prepared. Also, members of the public are free to direct any comments they may have on the Cost Study and Nonproliferation Impacts Assessment to DOE.

36-4: The current state of development of each treatment technology is described in Chapter 2 of the EIS. DOE recognizes that the treatment methods vary in their current state of development, and this was a factor in dismissing GMODS and the direct plasma arc-vitreous ceramic and chloride volatility processes from evaluation at this time. However, it was felt that the technologies analyzed in the EIS were developed to a sufficient level of maturity to permit consideration of their environmental impacts. It was not practical or necessary to wait until research on each technology has proceeded to a similar point prior to preparing the EIS. It should be noted that, under the option of continued storage under the No Action Alternative, the sodium-bonded spent nuclear fuel would continue to be stored safely until a less mature technology is developed to the point that it becomes a reasonable treatment alternative.

36-5: The commentor feels that DOE has not given other methods of treating sodium-bonded spent nuclear fuel the same consideration as electrometallurgical treatment. As stated in Section 1.3 of the EIS, as a

Commentor No. 36: David E. Adelman (Cont'd)

Reliance on other existing technologies or technology development programs must be adequately assessed in the EIS. Yet, in the Draft EIS only one of the six alternatives evaluated assesses a treatment method for the driver fuel other than the EMT technology. Further, this alternative ends up being the most costly because it involves melt-and-dilute treatment of the entire 60 tons of sodium-bonded fuel, as opposed to the 3 tons of driver fuel, and the treatment site is ANL-W, rather than the existing facility at the Savannah River Site ("SRS"). This illustrates a limitation of the current structure of the EIS. It would be more informative to provide environmental impact and cost information for the treatment of the two types of spent nuclear fuel separately. This would help identify the most environmentally protective (and cost effective) combination of treatment methods. 36-6

For example, DOE has not evaluated an alternative in which the blanket fuel is disposed of directly after removal of the bonded-sodium and the driver fuel is treated using the melt-and-dilute technology and facilities being developed at SRS.[3] Utilization of the existing SRS program would reduce costs by eliminating duplicative DOE research and development programs and facilities and benefit from economies of scale, which would in turn reduce their aggregate environmental impacts. The EIS must also include an evaluation of the reactivity of the sodium in the driver fuel and the potential risks this creates for its long-term disposal. It may be that the interdiffusion of the fuel, sodium, and cladding substantially reduces the reactivity of the sodium, making it acceptable for direct disposal in a geologic repository without further treatment. These alternatives, and their variants, must be evaluated in the EIS to arrive at the appropriate combination of treatment technologies for the driver and blanket fuel elements. 36-7 36-8 36-7 36-6 36-9 36-10 36-6

II. Conclusion

It is critical that DOE evaluate treatment strategies for the blanket and driver fuel elements separately. The EIS would benefit substantially from having the environmental and costs analyses of each treatment method presented separately for each fuel type. It is also essential that DOE evaluate alternatives that take advantage of existing technologies and programs, particularly the rapidly progressing melt-and-dilute project at SRS and well established mechanical methods for removing sodium from fuel elements. Finally, DOE cannot arbitrarily remove certain portions of its analysis, particularly its non-proliferation assessment, from the EIS; all aspects of DOE's assessment should be part of the EIS and available for public review and comment. 36-11 36-8 36-12 36-13

David E. Adelman
Project Attorney, Nuclear Program

[3] According to Natraj Iyer, the manager of the SRS melt-and-dilute program, "it is very realistic to make [the melt-and-dilute program a] success[]." He has also stated that they can allay the concerns that the Defense Nuclear Safety Board has raised.

Response to Commentor No. 36 (Cont'd):

result of comments received during the scoping period, DOE changed the proposed action of the EIS, the structure of alternatives, and the title of the EIS from the "Electrometallurgical Treatment of Sodium-Bonded Spent Nuclear Fuel in the Fuel Conditioning Facility at Argonne National Laboratory-West" to the "Treatment and Management of Sodium-Bonded Spent Nuclear Fuel." This change was made to address public concern about potential bias toward one treatment technology over others. The alternatives evaluated in the EIS were restructured to reflect differences in the characteristics of driver and blanket sodium-bonded spent nuclear fuel. Several alternatives were added to the EIS to address the treatment of driver and blanket spent nuclear fuel by different technologies. Conversely, because of the characteristics of sodium-bonded spent nuclear driver fuel, the maturity of existing technologies, and the availability of existing facilities to treat and manage the driver spent nuclear fuel, treatment technologies for driver spent nuclear fuel are currently limited to electrometallurgical and melt and dilute treatment technologies. A range of reasonable alternatives and technologies for the treatment of driver and blanket sodium-bonded spent nuclear fuel, as well as the No Action Alternative that includes direct disposal with no treatment, were evaluated in the EIS. In parallel, a separate assessment was conducted on the nonproliferation characteristics of all the treatment technologies considered in the EIS. The EIS and the conclusions of the Nonproliferation Impacts Assessment, along with other factors, will be considered during the decision-making process prior to publication of the Record of Decision.

36-6: As discussed in Section 2.5 of the EIS, although each alternative evaluates the treatment of both driver and blanket sodium-bonded spent nuclear fuel, the environmental impact analyses are sufficient to allow DOE to consider the separate treatment of driver and blanket fuel. As a result of the commentor's remarks, the possibility of treating sodium-bonded driver spent nuclear fuel using the melt and dilute process at the Savannah River Site was considered. It was dismissed from further evaluation, however, as indicated in the revised Section 2.6 of the EIS.

36-7: In response to public comments received at the public scoping meetings, DOE decided to analyze the driver and blanket spent nuclear fuel separately. Six treatment alternatives were evaluated in the EIS that included various combinations of fuel type and site location. However, as stated in Section 2.6 of the EIS, when preparing the Record of Decision DOE will consider all

Commentor No. 36: David E. Adelman

Response to Commentor No. 36 (Cont'd):

36-8: The SBSNF EIS was prepared in accordance with NEPA, Council on Environmental Quality regulations on implementing NEPA (40 CFR 1500-1508), and DOE's NEPA implementation procedures (10 CFR 1021). None of these require the inclusion of a cost analysis in an EIS. As discussed in the introduction, the basic objective of this EIS is to provide the public and DOE decision-makers with a description of the reasonable alternatives for the treatment and management of DOE's sodium-bonded spent nuclear fuel and their potential environmental impacts. However, DOE issued a separate Cost Study on August 12, 1999, that analyzes and compares the costs of the alternatives analyzed in the EIS. Cost will be considered during the decision-making process in preparing the Record of Decision.

36-9: Actual costs for treating and managing sodium-bonded spent nuclear fuel are not part of the EIS process. However, the cost of using SRS facilities is included in the August 1999 Cost Study. Cost will be one of the factors considered in preparing the Record of Decision for the treatment and management of sodium-bonded spent nuclear fuel.

36-10: As discussed in Section E.4.6, the EBR-II fuel at INTEC's Basins 666 and 66 are stored inside sealed stainless steel cans that prevent the contact of basin water with the fuel cladding. During the average 17 years of storage in Basin 666, 10 of the 2,148 cans were confirmed to have water in-leakage. With water inside these cans, a fuel-water reaction produced hydrogen gas, which created bubbles that allowed detection of the water in-leakage. These observations are consistent with the fact that sodium and metallic uranium react with water to produce hydrogen and this is the reason that all the sodium-bonded spent nuclear fuel is stored in dry storage or sealed containers that prevent the exposure of the fuel cladding to water. Under storage conditions in a geologic repository, fuel cladding could disintegrate over time, leading to the collection of a large amount of sodium within the confines of the storage can. If this fuel can were to fail, a large amount of sodium would be available to react with water in the repository. This could produce a violent reaction. DOE considers this condition to be unacceptable. The EIS, under the No Action alternative, analyzes a direct disposal option that is conditional on the acceptability of untreated sodium-bonded spent nuclear fuel in a repository. However, the feasibility and acceptability of such action remains to be determined.

36-11: Although each alternative presented in the EIS addresses the combined treatment and management of both driver and blanket sodium-bonded spent nuclear fuel, the analyses presented in Chapter 4 evaluate the impacts of the separate treatment of driver and blanket spent nuclear fuel. As

Commentor No. 36: David E. Adelman (Cont'd)

Response to Commentor No. 36 (Cont'd):

discussed in Section 2.5, DOE will consider the separate treatment of driver and blanket spent nuclear fuel in identifying a preferred alternative. In other words, DOE will consider combinations of technologies, options, and fuel types, including combinations not included among the specific combinations considered in the EIS.

36-12: The EIS evaluates reasonable treatment technologies (including existing technologies and programs) for the treatment and management of sodium-bonded spent nuclear fuel. The melt and dilute treatment process is part of Alternative 5, which is described in Section 2.5.6 of the EIS. The melt and dilute treatment process is also described in greater detail in Section 2.3.4 and Appendix C, Section C.5. The methods considered for removing metallic sodium from blanket sodium-bonded spent nuclear fuel elements are described in Section 2.3.9.

36-13: The Nonproliferation Impacts Assessment was prepared to provide additional pertinent information to the Secretary of Energy so that he may make an informed decision with respect to the treatment and management of DOE's sodium-bonded spent nuclear fuel. This document was mailed to interested parties on August 12, 1999, and was made available to attendees at all of the public hearings on the draft EIS. It should be noted that, although NEPA does not require inclusion of the information provided in the Nonproliferation Impacts Assessment in the EIS, it will be considered along with other pertinent data when the Record of Decision is prepared. Also, members of the public are free to direct any comments they may have on the Nonproliferation Impacts Assessment to DOE.

Commentor No. 37: Carol Murphy

SB SNF Toll Free Line

9/13/99

Carol Murphy
Dan Freeman

208-726-5929

I am calling to comment on the draft EIS. I am against pyroprocessing at the INEEL in Idaho and I'm also calling to request a 60 day extension. I understand that its been extended to September 28th but I believe it should be extended to at least the middle of November to get an EIS for Yucca Mountain and also until a full cost analysis has been done on different alternatives and several other reasons. I've written a letter. 37-1 37-2 37-3

Response to Commentor No. 37:

37-1: The commentor's objection to electrometallurgical treatment (pyroprocessing) of sodium-bonded spent nuclear fuel at INEEL is noted.

37-2: In an effort to ensure that all interested parties had time to comment on the draft EIS, the deadline for transmittal of comments was extended from September 13, to September 28, 1999 (64 FR 49169).

37-3: At the request of several members of the public, DOE prepared and issued a separate Cost Study during the public comment period on the draft EIS. Copies of the Cost Study were mailed to interested members of the public and were also available at the four public hearings during August 1999. The Yucca Mountain EIS was issued in July 1999.

Commentor No. 38: Suzy Nielond

SB SNF Toll Free Line

9/20/99

Suzy Nielond

307-739-2430

My comment is that I think you should extend the comment period. We need at least 60 days minimum to at least find out about this and get all the information about this before we decide that it's a bad idea which some of us have decided already. And that's my comment I'm calling from Jackson, Wyoming. 38-1

Response to Commentor No. 38:

38-1: In an effort to ensure that all interested parties had time to comment on the draft EIS, the deadline for transmittal of comments was extended from September 13 to September 28, 1999 (64 FR 49169). DOE made materials relevant to the review of the draft EIS available in public reading rooms and at a series of public hearings that were held on August 17, 1999, in North Augusta, South Carolina; August 24, 1999, in Boise, Idaho; August 26, 1999, in Idaho Falls, Idaho and August 31, 1999, in Arlington, Virginia. The materials placed in the reading rooms included the electrometallurgical demonstration project environmental assessment, the Finding of No Significant Impact for the environmental assessment, National Research Council interim status reports on the demonstration project, the 1995 Settlement Agreement and Consent Order with the State of Idaho, the EIS scoping meeting transcripts and public hearing comments, and the draft EIS hearing presentations and fact sheets. In addition, completion of the Cost Study and Nonproliferation Impacts Assessment was expedited so that they would be available to the public during the comment period. These reports were mailed to interested parties on August 12, 1999, and were made available to the public at the public hearings on the draft EIS. Although these reports are not critical to the environmental impact analysis presented in the EIS, they will provide input to the Record of Decision. While the final National Research Council report on the electrometallurgical treatment demonstration project at ANL-W was published in April 2000, interim status reports were produced throughout the project and this data was used to prepare the EIS, as discussed in Section 1.6.3 of the EIS.

Commentor No. 39: Carol Murphy and Dan Freeman

Carol Murphy
P.O Box 4714
Ketchum, ID.
83340

Dan Freeman
PO Box 4265
Ketchum, ID
83340

Ms. Susan Lesica
US Dept. of Energy
Office of Nuclear Facilities Mngmnt, NE-40
19901 Germantown RD.
Germantown, MD 20874-1290

Dear Ms. Lesica, Sept. 11, 1999

We are writing for two reasons.

One, is to ask for an extension on the comment period for the "DEIS for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel." We would like to see this comment period extended at least 60 days. We feel this is necessary because so much of the information relevant to the procedure is still unavailable. The public has the right to review this information, especially those of us in Idaho, therefore the comment period must be extended. PLEASE! 39-1 39-2

Secondly, we are writing to express our dissatisfaction with this proposed project for pyroprocessing. It is wasteful, takes money away from even greater environmental problems at the INEEL, and is just too efficient at "corporate welfare." Please DO NOT WASTE OUR TAXPAYER DOLLARS. Please, DO NOT CREATE MORE NUCLEAR WASTE IN IDAHO, OR ANYWHERE ELSE. 39-3 39-4 39-5

In summary, we are opposed to the procedure of pyroprocessing at the INEEL, and we feel that the comment period on the DEIS for this procedure needs to be extended for at least 60 days, and especially until the Yucca Mountain EIS is released.

Thank you for reading our comments.

Sincerely,

Carol Murphy
B.A. Economics
The Colorado College

Dan Freeman
B.A. Philosophy
Harvard University

Response to Commentor No. 39:

39-1: In an effort to ensure that all interested parties had time to comment on the draft EIS, the deadline for transmittal of comments was extended from September 13 to September 28, 1999 (64 FR 49169).

39-2: DOE made materials supporting preparation of the EIS available in the public reading rooms and at the public hearings held on August 17, 1999, in North Augusta, South Carolina; August 24, 1999, in Boise, Idaho; August 26, 1999, in Idaho Falls, Idaho; and August 31, 1999, in Arlington, Virginia. These materials included the environmental assessment for the Electrometallurgical Treatment Research and Demonstration Project, the Finding of No Significant Impact for the environmental assessment, National Research Council reports, the 1995 Settlement Agreement and Consent Order with the State of Idaho, scoping period meeting transcripts and comments, and the draft EIS hearing presentations and fact sheets. In addition, completion of the Cost Study and Nonproliferation Impacts Assessment was expedited so that they also would be available to the public during the comment period. These reports were mailed to interested parties on August 12, 1999, and were available at the public hearings on the draft EIS. Although these reports are not critical to the environmental impact analysis presented in the EIS, they will be considered during the decision-making process in the preparation of the Record of Decision. While the final National Research Council report on the Electrometallurgical Treatment Research and Demonstration Project at ANL-W was published in April 2000, interim status reports were produced throughout the project. Data generated during the demonstration project were used in preparing the EIS, is discussed in Section 1.6.3 of the EIS.

39-3: The commentor's objection to electrometallurgical treatment (pyroprocessing) of sodium-bonded spent nuclear fuel is noted.

39-4: Congress determines how funds are allocated. DOE spends monies consistent with Congressional direction. DOE is not in a position to make the difficult tradeoffs that may be required between alternative Federal programs and spending priorities. The issue of funding for the treatment and management of sodium-bonded spent nuclear fuel is beyond the scope of the SBSNF EIS.

39-5: Chapter 4 of the EIS presents data that demonstrates that, compared to leaving the sodium-bonded spent nuclear fuel in its current form, treatment and management of the sodium-bonded spent nuclear fuel would significantly reduce the volume of high-level radioactive waste that needs to be disposed of in a geologic repository.

Commentor No. 40: Julie Bowles

SB SNF Toll Free Line

9/27/99

Julie Bowles
7209 Valley Heights Drive
Boise, ID 83709

I am in favor of not doing the pryoprocessing and that I think that it's a cost issue. I think it's a health issue and I understand that the DOE is looking at not doing it at INEEL and I think that's the right way to go. 40-1,-2 40-3,-1

Response to Commentor No. 40:

40-1: The commentor's objections to electrometallurgical treatment (pyroprocessing) of sodium-bonded spent nuclear fuel at INEEL is noted. The commentor's support for other alternatives is also noted.

40-2: Actual costs for treating and managing sodium-bonded spent nuclear fuel are not part of the EIS process. However, the Cost Study shows that electrometallurgical treatment (pyroprocessing) of the sodium-bonded spent nuclear fuel is neither the most nor least expensive alternative. Information from the Cost Study, the EIS, the public comments, and other sources will factor into the decision-making process leading to the Record of Decision.

40-3: As indicated in the EIS, the human health effects resulting from operational activities to treat and manage sodium-bonded fuel are very small. The estimated cumulative health effects to the public residing in the vicinity of INEEL from current and reasonably foreseeable future activities are summarized in Section 4.11.1.4 of the EIS. As indicated in this section, the expected health effects from these activities are very small. For example, an individual residing at the INEEL site boundary would be expected to receive a maximum radiation dose of 0.065 millirem per year from all releases, compared to natural background doses of 360 millirem per year, and are well below the regulatory limit of 10 millirem per year. Appendix E, Section E.2.1, of the EIS provides the Federal and DOE regulatory limits on radiation exposures.

Commentor No. 41: Steve Hopkins (Cont'd)

FROM : FAX NO. : Sep. 28 1999 01:50PM P2

meaningful research to determine the effectiveness of pyroprocessing and thus the advisability of expanding the project. Now the DOE is moving forward with the EIS on expansion long before Argonne's demonstration project results have been analyzed. 41-4

Key areas of concern about pyroprocessing for both the public and Congress have always been its cost and its impact on US nonproliferation efforts. Though these separate documents were released during the public comment period, they were not initially available along side the DEIS, and having them as separate documents makes it harder for the public to review all the impacts associated with the various proposed treatments. The Alliance was told these documents would be mailed out on August 9th, however they were not made available to us until the Boise hearing on August 24th. Initially, this gave us only three weeks to review the documents, incorporate that information into information gathered from the DEIS, and pass the information onto our members. That is not nearly enough time and again speaks to the haste with which this process is proceeding. 41-5

Short-circuiting the nonproliferation analysis is particularly egregious in light of the pledge in the Notice of Intent to include this assessment in the draft EIS and the existence of such a DOE assessment from December 1998.

In 1994, then-DOE secretary Hazel O'leary asked Congress to stop funding the IFR. "Because it is based on plutonium reprocessing and recycle, continued development of the Integral Fast Reactor would undercut our efforts to discourage other countries from plutonium reprocessing and recycle." **Pyroprocessing represents the reprocessing component of the IFR program.** The fact that this technology could be, according to the DOE's recent non-proliferation assessment, at least perceived by other countries as a reprocessing technology for weapons material is compounded by INEEL's historical reprocessing role related to weapons production and the current on-site presence of plutonium and uranium suitable for bombs. 41-6

The National Academy of Science has regularly evaluated the pyroprocessing demonstration project, which has increased the scientific integrity of Argonne's project. But the NAS final report on pyroprocessing will not be complete until well after the EIS public comment period has ended, hampering both the public's ability to comment and Argonne's ability to evaluate its own work. 41-7

In 1995, Sandia National Laboratories recommended that "...**most decisions on [spent fuel] treatment or conditioning should wait until a repository type and site are known**" [bold italics in original]. Many observers, including the NAS, have repeatedly raised the issue of uncertainty vis a vis the waste forms that pyroprocessing will produce and their acceptability at a geologic repository. Since getting waste ready for a geologic repository is the justification for Argonne's project, it must not go forward until the waste produced by the demonstration project has been fully characterized, which will occur 41-8

Response to Commentor No. 41 (Cont'd):

electrometallurgical process alternatives was based on actual data from the project. Section 1.6.3 of the EIS summarizes the status and the results of the project.

41-5: In response to comments received during the scoping period, DOE expedited completion of the Cost Study and the Nonproliferation Impacts Assessment. These reports were mailed to interested parties on August 12, 1999, and also were made available to attendees at the public hearings on the draft EIS, which were held August 17, 1999, in North Augusta, South Carolina; August 24, 1999, in Boise, Idaho; August 26, 1999, in Idaho Falls, Idaho; and August 31, 1999, in Arlington, Virginia. NEPA does not require inclusion of the information presented in these documents in the EIS; however, it will be considered along with other pertinent data when the Record of Decision is prepared. DOE extended the comment period from September 13 to September 28, 1999 (64 FR 49169) to provide commentors with an additional two weeks to review the draft EIS and associated documents and to pass the information on to other interested parties.

41-6: Although the Nonproliferation Impacts Assessment is not part of the EIS process, it fully analyzes the potential nonproliferation impacts of each of the proposed alternatives and technologies addressed in the EIS. The Notice of Intent to prepare the EIS stated, "The combination of the information contained in the draft EIS, the public comments in response to the draft EIS, and the Nonproliferation Impacts Assessment will enable the Department to make a sound decision...." As stated in the Nonproliferation Impacts Assessment, the alternatives involving PUREX reprocessing and broad application of electrometallurgical treatment of both driver and blanket fuel have a greater potential to provide encouragement to other countries to engage in plutonium reprocessing. Given the small quantity and unique characteristics of the sodium-bonded spent nuclear fuel and the reason for its treatment, however, such encouragement, if any, would be limited. The proposed use of electrometallurgical treatment technology would not add to the stockpile of weapons-usable fissile materials.

41-7: While the final report on the Electrometallurgical Treatment Research and Demonstration Project from the National Academy of Science's National Research Council was not available to the public during the comment period on the draft EIS, interim status reports were available in the public reading rooms. Thus, the public had an opportunity to review the information

Commentor No. 41: Steve Hopkins

FROM : FAX NO. : Sep. 28 1999 01:49PM P1

Snake River Alliance

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September 28, 1999

Susan Lesica, Document Manager
Office of Nuclear Facilities Management
Office of Nuclear Energy, Science, and Technology
US Department of Energy, NE-40
19901 Germantown Road
Germantown, MD 20874-1290

Re: Comments of the Snake River Alliance on the "Draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel"

Dear Ms. Lesica,

The following comments are made on behalf of the 1,300 members of the Snake River Alliance, an Idaho-based grassroots group that has monitored activities at the Idaho National Engineering and Environmental Laboratory since 1990.

Thank you for the opportunity to comment on this DEIS and for extending the public comment period. In addition, we thank DOE for finalizing the blanket and driver fuel separately due to their different chemical and radiological properties, and for refocusing the scope of the project to reflect possible need for treatment as a waste as opposed to need for treatment by pyroprocessing. 41-1

As we previously stated during the scoping process on the Department of Energy's preparation of an environmental impact statement on electrometallurgical treatment (pyroprocessing) of sodium-bonded spent nuclear fuel, the preparation of this EIS is premature. In fact, it now should proceed with regarding the pyroprocessing of this special case spent fuel at this time. You will be spending your time doing so for reasons that clearly have nothing to do with protection of human health and the environment. 41-2

The fact that the DOE has reinstated funding to hold the Argonne pyroprocessing workforce in place between the start of FY2000 (for which no funding was previously requested) and the record of decision that will grow from the draft EIS, is ample grounds for suspecting that the budget process rather than sound science is the driving force behind this EIS process. A DOE source quoted in Nucleonics Week, June 8, 1995, admitted to as much when he described pyroprocessing at Argonne-West as "just about the only thing they have left to do. . . . It's a jobs issue." 41-3

During the environmental assessment process for the pyroprocessing demonstration project, Argonne argued that the number of blanket and driver spent fuel elements proposed for treatment through the EA was the absolute minimum required for 41-4

Response to Commentor No. 41:

41-1: The comment is noted. DOE revised the scope of the EIS based on comments provided during the public scoping period.

41-2: DOE is committed to improving its environmental management practices, to operating its facilities in a manner that meets or exceeds all applicable environmental, safety, and health requirements, and to the cleanup of its environmental problems. The focus of the EIS is to assess the potential environmental and health impacts associated with the treatment and management of sodium-bonded spent nuclear fuel. Although not final, the latest guidance provided by DOE's Office of Civilian Waste Management in their "Waste Acceptance System Requirements Document," Revision 3, April 1999 (see Section 4.12.1 of the EIS), indicates that it is highly probable that sodium-bonded spent nuclear fuel would not be acceptable in the repository without some stabilization and/or removal of the metallic sodium. The stabilization of the spent nuclear fuel and/or removal of the metallic sodium will provide greater protection of human health and the environment. In addition, having completed the Electrometallurgical Treatment Research and Demonstration Project (see Section 1.6.3) and in planning the closure of its PUREX processing capabilities, DOE needs to decide whether this process is suitable for treating the remaining sodium-bonded spent nuclear fuel, or whether there is sufficient reason to delay a decision and wait for the development of other treatment technologies. Delaying the EIS could result in the loss of capability and experienced, knowledgeable technical staff should DOE decide at a later date to use the electrometallurgical process to treat sodium-bonded spent nuclear fuel. Section 1.2 of the EIS has been revised for clarification.

41-3: ANL-W is involved in other DOE missions in addition to electrometallurgical treatment. Ongoing activities unrelated to electrometallurgical treatment at ANL-W include long-term waste storage gas generation testing at the Zero Physics Power Reactor; characterization and repackaging of mixed hazardous waste for shipment to the Waste Isolation Pilot Project at the Hot Fuel Examination Facility; conversion of sodium coolant from the EBR-II and Fermi reactors to chemically inert low-level radioactive waste in the sodium process facility; and deactivation of the EBR-II facility. The number of jobs affected by the electrometallurgical treatment alternative at ANL-W is presented in Section 4.2.3 of the EIS.

41-4: Final test results were made available in August 1999 and were used in the EIS. The success criteria established at the outset of the project were fulfilled. The environmental impact analysis associated with the

Commentor No. 41: Steve Hopkins (Cont'd)

FROM : FAX NO. : Sep. 28 1999 01:50PM P3

Though the Alliance's confidence in the governor's settlement agreement is not high, it should be noted that, if Idaho's sodium laced and/or bonded spent fuel is not reprocessed, the agreement mandates 2035 as its departure date. If it is reprocessed and therefore becomes high-level waste, there is no departure date in the agreement only a "target date" for road readiness. Argonne therefore has to account for long-term, not interim, storage costs (economic and environmental) at INEEL as well as all other costs associated with its production and disposition. 41-9

It's worth noting here that the proposed action for the 17 MTHM of EBR2 fuel at Savannah River is not pyroprocessing. 41-10

The Snake River Alliance has long supported the designation of spent fuel as a waste. The DOE has resisted that course. One result of this is that Argonne's 45% waste reduction argument for pyroprocessing is "off." At the very least, the DOE must account for the low enriched uranium pyroprocessing produces. To say it is a product, not a waste is to ignore the fact that, in the DOE's world, spent fuel is not a waste either. Discussion of the LEU stream must include a full analysis of what happens to this stream and when. 41-11

In the past, the Alliance has prevailed upon the DOE to include ANL-W as a part of INEEL in environmental analyses. We would now encourage the DOE to consider ANL-W a part of the DOE's complex itself. If this occurs, it would highlight the necessity for coordinating the analysis in the present draft EIS (and the action chosen in the ROD) with the EIS on spent fuel management at Savannah River and the EIS on stabilization of high-level waste at INEEL. The INEEL EIS has particular relevance here. If the two studies (and decisions) are not coordinated, there may well be three high-level waste forms in one Idaho county. Again, analysis of those waste forms requires a full accounting of the economic and environmental costs of long-term management at INEEL. 41-12 41-9

Summary:

According to the section on potential facility accidents (pg.4-11), the sodium-bonded fuel ". . . is in a very safe and stable configuration and no reasonable foreseeable accident scenarios could be identified." If the spent fuel in question presented a near term risk to human health and the environment, the Alliance would support an alternative that could best stabilize it to lessen that risk. It is also the case that if a repository were open (and waste acceptance criteria known), and if such a facility was supported by the public through an open and scientifically credible public process, treatment of this fuel could potentially be justified. However, the DOE has not made even a weak case for treating this fuel at this time; there is no current repository for this spent fuel (not even *known* waste acceptance criteria), and there may never be. These facts and our previously related concerns about the issuance of this DEIS before important data are in lead us to support the no action alternative at this time in accordance with the ROD for the Programmatic Spent Nuclear Fuel EIS. If a case can be made in the future for treatment of this spent fuel due to risks posed to human health and the environment, we ask the 41-13 41-14 41-15

Response to Commentor No. 41 (Cont'd):

demonstration project at ANL-W was published in April 2000. DOE will consider the data contained in this report in preparing the Record of Decision.

41-8: The process of establishing a repository is dependent on not only the site but also the materials to be disposed of. As part of most of the steps in this process a total system performance assessment that describes the probable behavior of a repository at Yucca Mountain is performed. The total system performance assessment includes the performance of the specific waste forms and inventories proposed for disposal. As part of this work to establish a repository, data for the waste forms are needed prior to final choice of the repository not after it. In fact, if specific waste forms are not represented in crucial documents like this EIS, additional documentation will be needed to allow the possibility of disposing of those materials in the repository. As part of the Electrometallurgical Treatment Research and Demonstration Project, ANL-W has interacted regularly with DOE and have provided conservative waste form data for the EIS.

41-9: This EIS evaluated the environmental impacts from treatment and management of sodium-bonded spent nuclear fuel up to 2035. This date is consistent with the State of Idaho Settlement Agreement and Consent Order that all spent nuclear fuel and high-level radioactive waste be removed from the State of Idaho by 2035. The commentor is correct in stating that the Settlement Agreement and Consent Order only requires the road-readiness of the high-level waste by the target date. Normal operation radiological effluent from potential fuel degradation during storage at INEEL up to 2035 are evaluated under the No Action Alternative in Section 4.2 of the EIS. As discussed in revised Section 2.5.1 of the EIS, a fundamental assumption made under the No Action Alternative is that sodium-bonded spent nuclear fuel would be disposed of in a repository, along with the rest of the DOE-owned spent nuclear fuel, within a finite period of time and under the institutional control of DOE. This SBSNF EIS covers a time period up to 2035, at which time sodium-bonded spent nuclear fuel stored in Idaho would need to be transported out of the state and either stored or treated at another DOE site. For such an eventuality, additional NEPA documentation would be required. The unlikely scenario that treated sodium-bonded spent nuclear fuel remains at its current site beyond 2035 because there is no geologic repository to accept it was evaluated as part of the No Action Alternative in Yucca Mountain Draft EIS, which was issued by DOE in July 1999. The Yucca Mountain EIS is discussed in Section 1.6.2.2.

41-10: EBR-II fuel currently located at SRS is declad blanket spent nuclear fuel that has been cleaned of sodium and placed in aluminum cans. This fuel is

Commentor No. 41: Steve Hopkins (Cont'd)

FROM : FAX NO. : Sep. 28 1999 01:51PM

DOE to reissue a Draft EIS incorporating the aforementioned information yet to be gathered, including a new nonproliferation assessment that assumes a more realistic view of pyroprocessing as a reprocessing technology. 41-15 (cont'd)

Again, thank you for the opportunity to comment on the draft EIS and for the deadline extension.

Respectfully submitted,

Steve Hopkins
Program Assistant
Snake River Alliance

Response to Commentor No. 41 (Cont'd):

not part of the sodium-bonded spent nuclear fuel considered in this EIS.

41-11: Section 4.1.2 and Section C.1 of Appendix C of the EIS describes the low enriched uranium product that would result from electrometallurgical treatment of sodium-bonded blanket spent nuclear fuel. After electrometallurgically treating the sodium-bonded spent nuclear fuel, metal ingots containing either low enriched or depleted uranium would be stored in the Materials Building within the Zero Power Physics Reactor at ANL-W, pending DOE's decision regarding final disposition of this uranium. Final disposition of the uranium product from electrometallurgical treatment is not within the scope of this EIS. DOE plans to conduct a separate NEPA review that will evaluate the disposition of surplus uranium.

41-12: As stated in the introduction, this SBSNF EIS follows the June 1995 Record of Decision (60 FR 28680) for DOE's Programmatic Spent Nuclear Fuel EIS, in which DOE decided to regionalize spent nuclear fuel management by fuel type for DOE-owned spent nuclear fuel. DOE also decided to: (1) continue environmental restoration activities at INEEL; (2) develop cost-effective treatment technologies for spent nuclear fuel and waste management; and (3) implement projects and facilities to prepare waste and treat spent nuclear fuel for interim storage and final disposition. The Record of Decision for the Programmatic Spent Nuclear Fuel EIS (60 FR 28680) provides the programmatic umbrella for the site-specific actions addressed in the EISs identified by the commentor, the SBSNF EIS, the Savannah River Spent Nuclear Fuel Management EIS, and the Idaho High-Level Waste and Facilities Disposition Draft EIS. As tiered NEPA documents, these EISs analyze the site-specific environmental impacts of implementing the actions proposed in each. The Savannah River Spent Nuclear Management Fuel EIS evaluates the impacts from the treatment of aluminum-clad and other spent nuclear fuel designated for treatment at SRS. The Idaho High-Level Waste and Facilities Disposition Draft EIS evaluates the impacts from processing specific amounts of calcined and sodium-bearing, high-level radioactive waste material currently located at INEEL. The materials (spent nuclear fuel and high-level radioactive waste) addressed in these EISs have unique characteristics and requirements which necessitate their separate evaluation. Each of the EISs identified by the commentor was incorporated by reference and used, as appropriate, in this SBSNF EIS. The contributory effects of these other ongoing NEPA actions at INEEL and SRS are evaluated as part of the cumulative impacts analysis for those sites (see

Commentor No. 41: Steve Hopkins

Response to Commentor No. 41 (Cont'd):

in Section 4.11.1.6 of the SBSNF EIS. DOE, in their Record of Decision, takes into account many factors besides this EIS, including ongoing DOE programs, missions, and related NEPA actions that have relevance (see Section 1.6 in the SBSNF EIS).

41-13: The timing for this action is a programmatic issue rather than a safety issue. As stated in Section 1.2 of the EIS, DOE considers it prudent to evaluate the alternative technologies now, while DOE is performing site characterization activities for the potential repository at Yucca Mountain. Although not final, the latest guidance provided by DOE's Office of Civilian Waste Management in their "Waste Acceptance System Requirements Document," Revision 3, April 1999 (see Section 4.12.1 of the EIS), indicates that it is highly probable that sodium-bonded spent nuclear fuel would not be acceptable in the repository without some stabilization and/or removal of the metallic sodium. The stabilization of the spent nuclear fuel and/or removal of the metallic sodium will provide greater protection of human health and the environment. Having completed the Electrometallurgical Treatment Research and Demonstration Project (see Section 1.6.3) and in planning the closure of its PUREX processing capabilities, DOE needs to decide whether these processes are suitable for treating the remaining sodium-bonded spent nuclear fuel, or whether there is sufficient reason to delay a decision and wait for the development of other treatment technologies. Delaying the EIS could result in the loss of capability and of experienced, knowledgeable technical staff should DOE decide at a later date to use the electrometallurgical process to treat the sodium-bonded spent nuclear fuel. Section 1.2 of the EIS has been revised for clarification.

41-14: The commentor's support for a No Action Alternative, under which the only activities taking place concerning sodium-bonded spent nuclear fuel would be those dictated by the Record of Decision for the Programmatic Spent Nuclear Fuel EIS, is noted.

41-15: As stated in the Nonproliferation Impacts Assessment, DOE's Office of Arms Control and Nonproliferation has determined that, for this specific application, electrometallurgical treatment of this spent nuclear fuel is fully consistent with U.S. policy with respect to reprocessing and nonproliferation since it does not separate plutonium for reuse. Plutonium would be part of the ceramic waste form, which is more resistant to plutonium recovery than metallic waste forms such as those resulting from the melt and dilute and high-integrity can alternatives.

Commentor No. 42: Margaret Macdonald Stewart

Forward Header

Subject: Comments on DEIS on treatment and mgmt of sodium-bonded spent nuclear fuel
Author: mstewart@snakeriveralliance.org_at_INTERNET
Date: 9/28/99 2:55 PM

COMMENTS BY MARGARET MACDONALD STEWART ON DRAFT EIS FOR TREATMENT AND MANAGEMENT OF SODIUM-BONDED SPENT NUCLEAR FUEL

Attention: Sue Lesica 28 September 1999
EIS Document Manager
Office of Nuclear Facilities Management
Department of Energy

The following are but a few of the comments I have to make on the Draft EIS on Treatment and Management of Sodium-Bonded Spent Nuclear Fuel:

To begin with, this DEIS is a very typical DOE document in that it is a magnificent example of "cart before the horse" mentality. The public is being asked to make comments on a document that is inadequate and is based on yet-unknown criteria. In order for people to make informed and intelligent comments, it would therefore make sense to allow them access to all pertinent information upon which to base those comments. 42-1

The comment period extension was not adequate. A 15-day extension is almost a joke considering the information made available by DOE is still not adequate. This extension period is a flimsy ploy to placate a public that has not been given all the information it needs to comment. 42-2

The DEIS is incomplete for the following reasons and more:

Where is the National Academy of Science's review of the proposed treatment? 42-3

The cost analysis of various alternative treatment methods is missing. 42-4

This entire document is based on transporting the pyroprocessing waste to Yucca Mountain. The Yucca Mountain EIS has not yet been released and no one knows the waste acceptance criteria. It could well end up that the pyro waste would be unacceptable for Yucca Mountain storage. What would be the end scenario in such an event? 42-5

I have not received a copy of the recently released DOE nuclear weapons proliferation assessment. Why? Has it really been released? How do I get a copy, since I receive nearly everything DOE ever prints? 42-6

Response to Commentor No. 42:

42-1: DOE has made every effort to obtain and analyze all of the information it needs to make a decision on the treatment and management of its sodium-bonded spent nuclear fuel. DOE has analyzed input from the public (during the public scoping and comment periods on the draft EIS), as well as from Federal and state agencies and local and Tribal governments. It has also reviewed site-specific information on the environmental conditions prevailing at ANL-W, INEEL, and SRS, as well as documentation related to each of the proposed treatment technologies. DOE made material supporting the preparation of the EIS available in public reading rooms and at a series of public hearings that were advertised in the Federal Register, as well as local newspapers. In addition, completion of the Cost Study and Nonproliferation Impacts Assessment was expedited so that they would be available to the public during the comment period. These reports were mailed to interested parties on August 12, 1999, and were made available to attendees at all of the public hearings on the draft EIS, which were held August 17, 1999, in North Augusta, South Carolina; August 24, 1999, in Boise, Idaho; August 26, 1999, in Idaho Falls, Idaho; and August 31, 1999, in Arlington, Virginia. While the final National Research Council report on the demonstration project at ANL-W was published in April 2000, interim status reports were produced throughout the project and are available in the public reading rooms. Considering the additional time provided by the extension of the comment period and the availability of the data used to prepare the EIS, DOE does not believe that a second draft is warranted.

42-2: The original comment period on the draft EIS was set at 45 days in compliance with the Council on Environmental Quality's "Regulations for Implementing the Procedural Provisions of the National Environmental Policy Act" (40 CFR 1506.10(c)). In an effort to ensure that all interested parties had time to comment on the draft EIS, the deadline for transmittal of comments was extended from September 13 to September 28, 1999 (64 FR 49169). The extension of the comment period reflects DOE's commitment to the NEPA process by ensuring that the public had more time to review the EIS than the 45-day period required by Council on Environmental Quality guidelines.

42-3: The National Academy of Sciences' National Research Council prepared interim status reports on the results of the Electrometallurgical Treatment Research and Demonstration Project that have been reviewed by DOE.

Commentor No. 42: Margaret Macdonald Stewart (Cont'd)

The test or demonstration project on the proposed treatment of this sodium-bonded spent nuclear fuel being done at Argonne-West will not be completed with results available until the end of September at the earliest. How can the public comment on this proposed treatment without the results of this critical demo project??

42-7

I respectfully request that the DOE complete all the necessary documentation and resubmit a second DEIS on this subject, when it has included ALL the information necessary for the public to make informed and accurate comments. With only the information made available in this inadequate DEIS, it is obvious that DOE has already made up its mind on the preferred alternative, and this leaves the public - once again - feeling they have been left out of the democratic process when it comes to decisions made by DOE. A very poor attempt that needs to be redone.

42-8

42-9

42-8

Sincerely,

Margaret Macdonald Stewart
Box 4090
Ketchum, ID 83340

Response to Commentor No. 42 (Cont'd):

treatment demonstration project in September 1999 and published their final summary report in April 2000. The final report findings will be considered during the decision-making process leading to the Record of Decision.

42-4: The SBSNF EIS was prepared in accordance with NEPA, Council on Environmental Quality regulations on implementing NEPA (40 CFR 1500-1508), and DOE's NEPA implementation procedures (10 CFR 1021). None of these require the inclusion of a cost analysis in an EIS. As discussed in the introduction, the basic objective of this EIS is to provide the public and DOE decision-makers with a description of the reasonable alternatives for the treatment and management of DOE's sodium-bonded spent nuclear fuel and their potential environmental impacts. However, DOE issued a separate Cost Study on August 12, 1999, that analyzes and compares the costs of the alternatives analyzed in the EIS. Cost will be considered during the decision-making process in preparing the Record of Decision.

42-5: As discussed in Section 2.7 of this EIS, final waste acceptance criteria are still being developed for a geologic repository. DOE expects the waste forms that would be produced by the proposed action would be suitable for disposal in a geologic repository. In July 1999, DOE published a Draft Yucca Mountain EIS, which is discussed in Section 1.6.2.2 of this EIS. The Yucca Mountain EIS assumes that sodium-bonded spent nuclear fuel is treated using the electrometallurgical process prior to emplacement in the repository.

42-6: The Nonproliferation Impacts Assessment was mailed to those persons on the SBSNF EIS mailing list on August 12, 1999. It was also made available to attendees at the public hearings on the draft EIS, which were held August 17, 1999, in North Augusta, South Carolina; August 24, 1999, in Boise, Idaho; August 26, 1999, in Idaho Falls, Idaho; and August 1, 1999, in Arlington, Virginia. A copy of the report has been forwarded to the commentor.

42-7: Final test results were made available in August 1999 and were used in the EIS. The success criteria established at the outset of the project were fulfilled. The environmental impact analysis associated with the electrometallurgical process alternatives was based on actual data from the project. Section 1.6.3 of the EIS summarizes the status and the results of the project.

42-8: DOE has made material supporting the preparation of the EIS available in public reading rooms and through a series of public hearings which were advertised in the Federal Register, as well as local newspapers. In addition,

Commentor No. 42: Margaret Macdonald Stewart

Response to Commentor No. 42 (Cont'd):

completion of the Cost Study and Nonproliferation Impacts Assessment was expedited so that these would be available to the public during the comment period. These reports were mailed to interested parties on August 12, 1999, and were made available to attendees at all of the public hearings on the draft EIS. While the final National Research Council report on the Electrometallurgical Treatment Research and Demonstration Project at ANL-W was published in April 2000, interim status reports have been produced throughout the project and these are available in the public reading rooms. Considering the additional time provided by the extension of the comment period and the availability of the data used to prepare the EIS, DOE does not feel that a second draft is warranted.

42-9: The NEPA process provides a number of opportunities for the public to participate in the preparation of an EIS. For example, the public had the opportunity to attend scoping meetings and public hearings on the draft EIS, at which time they could make comments and speak directly to DOE and ANL personnel. These meetings were held in North Augusta, South Carolina; Boise, Idaho; Idaho Falls, Idaho; and Arlington, Virginia. The public also had the opportunity to comment on the EIS through the U.S. mail, e-mail, a toll-free FAX number, and a toll-free phone number. DOE takes this participation seriously. For example, DOE made a number of changes in the draft EIS in response to comments received during the scoping meetings, including dropping electrometallurgical treatment of sodium-bonded spent nuclear fuel at ANL-W as the Preferred Alternative from the beginning of the EIS process. In preparing the final EIS, DOE also carefully considered all comments received from the public. Thus, the public was not left out of the NEPA process for preparing this EIS.

Commentor No. 43: Willie R. Taylor

United States Department of the Interior

OFFICE OF THE SECRETARY
Washington, D.C. 20240

1849•1999

In Reply Refer To:
ER 99/637

SEP 7 1999

Ms. Susan Lesica
Document Manager
Office of Nuclear Facilities Management (NE-40)
Office of Nuclear Energy, Science, and Technology
U. S. Department of Energy
19901 Germantown Road
Germantown, MD 20874-1290

Dear Ms. Lesica:

The United States Department of the Interior (Department) has reviewed the draft environmental impact statement (DEIS) for the treatment and management of sodium-bonded spent nuclear fuel and offers the following comments.

General Comments

From review of the DEIS it is unclear if the proposed treatment and storage of sodium-bonded spent nuclear fuel at the Savannah River Site will represent a substantial risk to Departmental trust resources. The analysis reveals that all alternatives will impact air quality and at least one of the alternatives will impact water resources. It has also accordingly been shown that ecological receptors occur within the affected environment. However, the document does not address potential impacts to these resources in the discussion of environmental consequences. We suggest that further revisions of the EIS reflect that appropriate consideration was given to both the human and ecological environments. 43-1

Specific Comments

Section 2.10.2 -The second sentence of the second paragraph, states the radiological and nonradiological gaseous and liquid effluents, as well as the associated exposures to workers and the public, are well below regulatory standards and guidelines. However, these referenced standards and guidelines are not clearly represented in the document. It would be helpful if they were incorporated into *Table 2-4 Summary of Environmental Consequences for the Treatment and Management of Sodium-bonded Spent Nuclear Fuel* in the revised EIS. 43-2

Section 4.4.4.1. - In the last paragraph of this section there is a typo which references a section of the text that does not exist, 3.2.12.2. We believe that the proper section identifier is 3.2.10.2. 43-3

Response to Commentor No. 43:

43-1: As stated in Section 4.1.1, no radiological damage to plant and animal populations would be expected as the result of the proposed action because the estimated doses to the human population are well below threshold values for which effects to plants and animals would be expected. The EIS also identifies chemical releases to the air and water resources at SRS. These releases are essentially independent of the fuel being processed. They are generated from the operations of various facilities. The quantities of releases attributable to treatment of the fuel in this EIS are a very small fraction of the current releases at the site. Recent site environmental reports (years 1996-1998) did not identify any measurable impacts on plants and animals because the amounts emitted are very low or the chemicals have little potential for causing negative effects. Therefore, no chemical damage to plant and animal populations are expected to result from treatment of the fuel, as explained in this EIS.

43-2: Regulatory limits and guidelines for radiological and nonradiological effluent and associated exposures to workers and members of the public are presented in Section 4.1.3 and Appendix E of the EIS. Appropriate footnotes have been added to Table 2-4.

43-3: The commentor is correct. The section numbering cited by the commentor has been revised.

Commentor No. 43: Willie R. Taylor (Cont'd)

Section 4.10.2.1 - Assessment of impacts to air resources from nonradiological and radiological air pollutants is limited to concentrations at the site boundary. Many fish and wildlife resources, and possibly humans, will likely be exposed to air pollutants within the site boundary. Therefore, it is likely that impacts from air pollutants have been underestimated for wildlife populations occurring within the site boundary. 43-4

The Department appreciates the opportunity to review the DEIS. We hope our comments will be useful in your evaluation of various alternatives for this project. We will be happy to provide any further assistance that you may need.

Sincerely,

Willie R. Taylor
Director
Office of Environmental Policy
and Compliance

Response to Commentor No. 43 (Cont'd):

43-4: Site annual environmental reports monitor conditions within the site boundaries at SRS and INEEL and have not identified any measurable impacts on fish and wildlife resources. Releases and emissions as a result of the proposed action are a small fraction of the current releases and emissions from each site. Therefore, no impacts to ecological resources are expected to occur from the incremental contribution to cumulative impacts at SRS or INEEL from the treatment and management of sodium-bonded spent nuclear fuel.

Commentor No. 44: Kathryn Graves

Kathryn Graves
Box 4185
Hailey, ID 83333

Ms Susan Lesica
US Dept. Energy
Office of Nuclear Facilities Mgt., NE-40
19901 Germantown Rd.
Germantown, Maryland
20874-1290

September 7, 1999

To whom it may concern:

I would like to comment on the "Draft Environmental Impact statement for the Treatment and Management of Sodium-Bonded spent Nuclear Fuel."

I think there are several things wrong with pyroprocessing. 44-1

Firstly, it separates out bomb-grade uranium. Isn't that counterproductive to our nonproliferation goals?

Secondly, this will create new forms of nuclear waste. We have thirty to forty years worth of nuclear waste here at INEEL that we don't know what to do with. 44-2

Thirdly, there are great waste/storage/and Clean-up problems at INEEL that need to be dealt w/ first. 44-3

Please note that I am against pyroprocessing. And I am also requesting a 60-day extension of the comment period. 44-4

Sincerely,
KGraves

Response to Commentor No. 44:

44-1: The assessment of nonproliferation impacts is not a part of the EIS process. None of the alternatives analyzed in this EIS would generate weapons-usable fissile materials at INEEL. Although highly enriched uranium would be an interim product, it would be down-blended to low enriched uranium during electrometallurgical treatment.

44-2: Electrometallurgical treatment (pyroprocessing) has been evaluated and successfully demonstrated in a three-year program at ANL-W that was continuously reviewed by a National Academy of Sciences' National Research Council Committee that concluded that electrometallurgical treatment is a feasible process for treating sodium-bonded spent nuclear fuel. All of the alternatives evaluated in this EIS would produce some forms of high-level radioactive waste. The electrometallurgical treatment alternative produces two new waste forms, both of which are more stable than nontreated sodium-bonded spent nuclear fuel. DOE is confident that these new waste forms will be acceptable for emplacement in a geologic repository. All waste, storage, and cleanup problems are being addressed in parallel with the SBSNF EIS. Other EISs that have been or are expected to be issued evaluate radioactive waste, and spent nuclear fuel at INEEL.

44-3: The commentor's objection to electrometallurgical treatment (pyroprocessing) of sodium-bonded spent nuclear fuel is noted.

44-4: In an effort to ensure that all interested parties had time to comment on the draft EIS, the deadline for transmittal of comments was extended from September 13 to September 28, 1999 (64 FR 49169).

Commentor No. 45: Marlise A. Teasley

Sept 2-99

Ms. Dawn Lewis
US DOE
Office of Nuclear Facilities Mgt. NE40
19901 Germantown Road
Germantown, Maryland 20874-1290

Dear Ms. Lewis:

I am a native Idahoan who has followed as closely as I could the many problems at the INEEL for the past 15 years.

I have visited the site two times and there, have some idea of the magnitude of the waste management problems created in the past which are compounded by the continuing use of this already overburdened facility as a "waste dump."

Because of the lack of visible information regarding the feasibility of reprocessing, the cost, the

45-1

45-2

Response to Commentor No. 45:

45-1: DOE is committed to improving its environmental management practices, to operating its facilities in a manner that meets or exceeds all applicable environmental, safety, and health requirements, and to the cleanup of its environmental problems. DOE has a very aggressive cleanup program and has worked with the EPA, states, and stakeholders to develop long-range programs and commitments to clean up its facilities to acceptable levels. As stated in the introduction to this EIS, DOE proposes to treat the sodium-bonded spent nuclear fuel and facilitate its ultimate disposal in a geologic repository outside the State of Idaho. While the commentor's opinion about INEEL is noted, this comment is beyond the scope of the SBSNF EIS. The focus of the SBSNF EIS is to assess the potential environmental and health impacts associated with the treatment and management of sodium-bonded spent nuclear fuel.

45-2: As stated in the introduction to the SBSNF EIS, the programmatic risk in implementing any of the potential alternatives for the treatment and management of sodium-bonded spent nuclear fuel, or of not treating this fuel, is the uncertainty surrounding the acceptability of DOE spent nuclear fuel for emplacement in a potential geologic repository. Although not final, the latest guidance provided by DOE's Office of Civilian Waste Management in their "Waste Acceptance System Requirements Document," Revision 3, April 1999 (see Section 4.12.1 of the EIS), indicates it is highly probable that sodium-bonded spent nuclear fuel would not be acceptable in the repository without some stabilization and/or removal of the metallic sodium. The stabilization of the spent nuclear fuel and/or removal of the metallic sodium will provide greater protection of human health and the environment. Having completed the Electrometallurgical Treatment Research and Demonstration Project (see Section 1.6.3), and in planning the closure of its PUREX processing capabilities, DOE needs to decide whether these processes are suitable for treating the remaining sodium-bonded spent nuclear fuel, or whether there is sufficient reason to delay a decision and wait for the development of other treatment technologies. Delaying the EIS could result in a loss of capability and of experienced, knowledgeable technical staff should DOE decide at a later date to use the electrometallurgical process to treat sodium-bonded spent nuclear fuel. Section 1.2 of the EIS has been revised for clarification.

Commentor No. 45: Marlise A. Teasley (Cont'd)

of use in Idaho, as well as those in our neighboring states (Wyoming - including Jerry Obrea - + Montana in particular) plus the lack of actual waste acceptance criteria for the Yucca Mountain Project - which must be done before any action on prepresenting at all. I contend the DOE is not in any position, at this point in time, to move forward on their proposed solution to our waste management woes.

Certainly, the first step in this whole issue is to allow public comment for a minimum of 60 days beyond the September 13, 1999 deadline.

Let's not be guilty of haste in a decision which may have untold + far-reaching adverse effects for years to come.

Respectfully,

Marlise A. Teasley Ed.S.

610 Sunview Dr. Twin Falls, Ida. 83301

45-2 (cont'd)

45-3

45-2

Response to Commentor No. 45 (Cont'd):

45-3: In an effort to ensure that all interested parties had time to comment on the draft EIS, the deadline for transmittal of comments was extended from September 13 to September 28, 1999 (64 FR 49169).

Commentor No. 46: H. Zerriffi & L. Ledwidge

Sept. 29, 1999

To: Susan Lesica, NE-40
Fr: Lisa Ledwidge
cc: Hisham Zerriffi

Attached please find an UPDATED VERSION of IEER's comments on the Draft
Environmental Impact Statement for the Treatment and Management of
Sodium-Bonded Spent Nuclear Fuel.

We noticed a slight grammatical error on the version that was emailed to
you yesterday (from Hisham Zerriffi). Please replace that version with the
one herein.

Thank you very much.

Lisa Ledwidge
Outreach Coordinator and Editor, Science for Democratic Action
Institute for Energy and Environmental Research (IEER)
6935 Laurel Ave., Suite 204
Takoma Park, MD 20912 USA
(301) 270-5500 fax: (301) 270-3029
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SBSNFCOM.DOC

Response to Commentor No. 46:

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Comments of the Institute for Energy and Environmental Research (IEER) on the *Draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel* (DOE/EIS-0306D)

Hisham Zerriffi, Project Scientist
Lisa Ledwidge, Outreach Coordinator
9/28/1999

The Department of Energy (DOE) has once again released a Draft Environmental Impact Statement which demonstrates its willingness to sacrifice environmental protection and nonproliferation in order to pursue its programmatic goals. The *Draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel* (DOE/EIS-0306D, hereinafter referred to as DEIS) analyzes options for processing spent nuclear fuel which is contaminated with metallic sodium. With the exception of the "No Action Alternative" all six alternatives involve processing the spent fuel in such a way as to have serious repercussions on the proliferation of plutonium. 46-1

What is worse is that this document makes it clear that the DOE wants to undertake a proliferation risky program under the guise of Environmental Management. And, to make matters even worse, one of the technologies being proposed is responsible for a significant portion of the Energy Department's environmental management problems. PUREX is a technology which for decades has separated plutonium for use in nuclear weapons and created extensive environmental problems and safety risks in both Washington and South Carolina. There are currently millions of gallons of liquid radioactive waste from PUREX operations threatening both the Columbia River and the Savannah River. 46-2

The Energy Department proposes six alternatives for treating spent nuclear fuel contaminated with reactive metal sodium. The fuel is the result of the research program attempting to develop liquid metal fast breeder nuclear reactors. These used liquid metal sodium as a coolant.[1] The spent fuel consists mainly of two types: driver (used to maintain the chain reaction in the core) and blanket (used to breed plutonium-239). During irradiation, some of the metallic sodium enters and mixes with both the fuel and outer cladding (in the case of the driver fuel) or mixes with the outer cladding (in the case

[1] See Arjun Makhijani and Scott Saleska, *The Nuclear Power Deception: U.S. Nuclear Mythology From Electricity "Too Cheap to Meter" to "Inherently safe" Reactors.* (New York: Apex Press, 1999) for a description of fast breeder reactors and an explanation of why metallic sodium was used as a coolant.

46-1: Although the assessment of nonproliferation impacts is not part of the EIS process, DOE's Office of Arms Control and Nonproliferation assessed the potential nonproliferation impacts that may result from each of the proposed alternatives and technologies analyzed in this EIS. The report stated that, for this specific application, all of the alternatives except PUREX processing at SRS are fully consistent with U.S. policy on reprocessing and nonproliferation. Alternative 3, PUREX processing, is the only alternative that would generate weapons-usable fissile material, including plutonium. This plutonium would be managed along with other surplus plutonium, as described in the Surplus Plutonium Disposition EIS.

46-2: As described in Section 2.5.4 of the EIS, DOE is considering PUREX processing at F-Canyon as one of the alternatives for treatment and management of the sodium-bonded spent nuclear fuel. This process, as explained in Section 4.5.6, would produce liquid high-level and low-level radioactive waste. The liquid high-level radioactive waste would be vitrified at the Defense Waste Processing Facility and transformed to a borosilicate glass waste form in preparation for disposal in a geological repository. DOE has evaluated the impacts from current and future liquid waste storage and processing in the Defense Waste Processing Facility EIS and its Supplement (DOE/EIS-0082 and DOE/EIS-0082-S), as well as the Interim Management of Nuclear Material EIS (DOE/EIS-0220). Section 3.3.4.1 of this EIS and annual SRS environmental reports provide descriptions of current water quality conditions in the Savannah River at SRS. The liquid radiological effluent from PUREX treatment of declad and cleaned blanket spent nuclear fuel in F-Canyon would not exceed current operating parameters. The impacts of processing the liquid radioactive waste currently stored at the Hanford, Washington, site are beyond the scope of this EIS.

Commentor No. 46: H. Zerriffi & L. Ledwidge (Cont'd)

of blanket fuel). (p. 2-1 – 2-3) Of the 60 metric tons of sodium-bonded spent fuel, 57 metric tons is blanket fuel which can have the sodium removed through a method which does not process the fuel element itself (other than to remove the outer cladding).[2]

It is necessary to begin by noting the fact that this action is premature. The DEIS identifies no immediate environmental or health concerns for this action. Rather, the Purpose and Need for Action is because "[T]he presence of metallic sodium in the sodium-bonded spent nuclear fuel could complicate the disposal certification and licensing for the ultimate disposal of this spent nuclear fuel in a geologic repository." (p.1-3) This is because metallic sodium can react violently under certain conditions, including a possibility of spontaneous ignition. These reactions produce "heat, potentially explosive hydrogen gas, and sodium hydroxide, a corrosive substance." (p. 1-3) However, these concerns relate to the long-term management of this spent fuel. As is indicated in the "No Action Alternative" it is possible to undertake some stabilization activities to prevent immediate degradation while other alternatives are developed.

The DEIS goes on to state that pure plutonium metal and pure uranium metal are also reactive and the "waste acceptance criteria probably will exclude reactive materials unless their packaging minimizes the probability of rapid oxidation." (p. 1-3) Finally, the DEIS notes that some of the fuel contains HEU which "may require special criticality control measures." (p. 1-3) 46-3

However, some very obvious facts concerning the purported "Need for Action" call into question both the need and timing of this action.

- There is no pressing need for this action in order to address immediate safety, environmental or health problems. This needs to be clearly stated in the "Purpose and Need for Action" section of the final Environmental Impact Statement. What is at issue is the "disposal certification and licensing for the ultimate disposal of this spent fuel in a geologic repository." (1-3) The spent fuel can be managed with better storage after some minimal preparation.
- There is no guarantee that Yucca Mountain will be selected as the high-level waste repository. Considerable technical controversy over its suitability still remains. 46-4
- Even if Yucca Mountain is chosen, the final waste acceptance criteria have not yet been established and the DEIS itself states that there is a programmatic risk that the final waste forms will not meet the criteria. (1-1) The argument in the DEIS that potential waste forms should be developed in parallel with the repository is inconsistent with the fact that processing would start in the Year 2000. This is five years before the estimated time for receiving a construction permit from the Nuclear Regulatory Commission, a necessary step in developing final waste form criteria. The DOE is proposing to actually process this spent fuel, not develop "potential waste 46-5 46-6

[2] All six alternatives involve processing of at least a portion of the spent fuel. Five of the six alternatives would use a process called Electrometallurgical Treatment (EMT) at Argonne National Laboratory – West at the Idaho National Engineering and Environmental Laboratory in Idaho. This process, a subset of a reprocessing technology called pyro-processing, entails significant proliferation risks which are discussed below. Additionally, one of the options would use the traditional PUREX reprocessing technology at the Savannah River Site in South Carolina for a portion of the fuel.

Response to Commentor No. 46 (Cont'd):

46-3: The timing for this action is a programmatic issue rather than a safety issue. As stated in Section 1.2 of the SBSNF EIS, DOE considers that it is prudent to evaluate the alternative technologies now, while DOE is performing site characterization activities for the potential repository at Yucca Mountain. Although not final, the latest guidance provided by DOE's Office of Civilian Waste Management in their "Waste Acceptance System Requirements Document," Revision 3, April 1999 (see Section 4.12.1 of the EIS), indicates that there is a high probability that sodium-bonded spent nuclear fuel would not be acceptable in the repository without some stabilization and/or removal of the metallic sodium. The stabilization of the spent nuclear fuel and/or removal of the metallic sodium will provide for a greater protection of human health and the environment. Having completed the Electrometallurgical Treatment Research and Demonstration Project (see Section 1.6.3) and in planning the closure of the PUREX processing capabilities, DOE now needs to decide whether these processes are suitable for treating the remaining sodium-bonded spent nuclear fuel, or whether there is sufficient reason to delay a decision and wait for the development of other treatment technologies. Delaying the EIS could result in the loss of capability and experienced, knowledgeable technical staff should DOE decide at a later date, to use the electrometallurgical process to treat sodium-bonded spent nuclear fuel. Section 1.2 of the EIS has been revised for clarification.

46-4: The commentor's support for continued storage is noted. The SBSNF EIS does not assume that Yucca Mountain will be selected as the high-level waste repository. It only assumes that, at some time in the future, a geologic waste repository will be licensed and operated by DOE which would receive spent nuclear fuel and high-level radioactive waste.

46-5: See response to comment 46-3.

46-6: DOE acknowledges the commentors' support of the No Action Alternative. As stated in the introduction to the EIS, the programmatic risk in implementing any of the potential alternatives for the treatment and management of sodium-bonded spent nuclear fuel, or of not treating this fuel, is the uncertainty surrounding the acceptability of DOE spent nuclear fuel for placement in a potential geologic repository. (See response to comment 46-3.) The development of waste forms in parallel with the development of the repository is one of many considerations discussed in

forms." Therefore, these are not parallel processes, but rather sequential processes undertaken in the wrong order. 46-6 (cont'd)

- There are no immediate time constraints posed by the State of Idaho Settlement Agreement since it does not require spent fuel to be removed until 2035 and the longest processing time listed in the DEIS is 13 years. 46-7
- Of the 60 MT of spent fuel, 57 MT can have the sodium removed without any of the processing listed in the DEIS through the high temperature cleaning process described in Section 2.4.9. This fact should be made much clearer in the DEIS. It should also note that the uranium in this 57 MT is not HEU. Therefore, if the final waste acceptance criteria require sodium removal, it would only apply to 3 MT of fuel. 46-8
- The DEIS does not mention the fact that there is significantly more HEU in naval spent nuclear fuel slated to go to Yucca Mountain than is contained in the sodium bonded spent fuel.[3] As the DOE proposes to directly place naval fuel in Yucca Mountain, it is not clear why the HEU in the sodium fuel needs to be separated and isotopically diluted. 46-9
- The DEIS does not mention the fact that the sodium bonded spent fuel is only a small portion of the metallic uranium fuel in the Energy Department's inventory.[4] The majority of the Uranium metal fuel (including fuels made of uranium alloys) is from the N-reactor. There is no indication that N-reactor fuel will be processed to change the form of the Uranium. As this fuel will also contain plutonium, it is not clear why the metal form of the U and Pu in the sodium bonded spent fuel is a problem that cannot be mitigated. The Environmental Impact Statement itself states that reactive metals may be allowed if packaged to minimize rapid oxidation. (1-3) The overwhelming emphasis placed on the sodium in the DEIS also indicates that the metal U and Pu are of secondary concern. If this is not the case, the DEIS should be clear under what circumstances the U or Pu can be expected to be reactive and what other mitigation measures are possible. 46-10
- The DEIS should also be clear that some of the processing is specifically for the metallic uranium and plutonium. For example, PUREX processing of blanket fuel would occur *after* sodium removal in an argon hot-cell. Thus, PUREX is not a factor in making the fuel safe from the risks of metallic sodium. PUREX would be used to change the form of the uranium (the final form of the plutonium would still be metal, only it would be separated plutonium metal). 46-11

In the "Background" section of the DEIS, it is noted that the research and demonstration project for Electrometallurgical Treatment (EMT) was coming to an end in August, 1999. EMT is one of the main technologies under consideration in this DEIS, and, in fact, the original scope of the DEIS was supposed to be only EMT. What the 46-12

[3] The DOE anticipates 65 metric tons of heavy metal (MTHM) of naval spent nuclear fuel through 2035. U.S. Department of Energy, Office of Civilian Radioactive Waste Management, *Draft Environmental Impact Statement for a geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada.* DOE/EIS-0250D, July 1999. p. A-24

[4] ibid., p. A-24

consideration is the removal or conversion of metallic sodium to a nonreactive form.

46-7: The timing for the proposed action is not primarily dictated by constraints imposed by the State of Idaho Settlement Agreement and Consent Order. See response to comment 46-3.

46-8: The EIS, under Alternative 2 (Section 2.5.3), analyzes the environmental impacts of removing sodium from 57 metric tons of blanket sodium-bonded spent nuclear fuel and the subsequent packaging of this fuel in high-integrity cans. The environmental consequences of this action are presented in Section 4.4. As described in Appendix D, Section D.3.2.2, the uranium in the 57 metric tons of blanket fuel is depleted uranium and not highly-enriched uranium. Section 2.2 of the EIS was revised to be consistent with the information presented in Appendix D. If the finalized waste acceptance criteria for the repository require the removal of sodium from the spent nuclear fuel, this requirement would apply to all of the 60 metric tons of sodium-bonded spent nuclear fuel addressed in this EIS. As described in Sections 2.2 and 2.3.9 (Section 2.4.9 in the draft EIS), different treatment methods are required for the removal of sodium from driver fuel (3 metric tons) and blanket fuel (57 metric tons).

46-9: Disposal of HEU requires criticality control measures. Isotopic dilution of the HEU, while not necessary, would alleviate criticality concerns.

46-10: Section 2.2 of the EIS states that the 60 metric tons of heavy metal of sodium-bonded spent nuclear fuel constitutes approximately 2 percent of DOE's total current spent nuclear fuel inventory of nearly 2,500 metric tons of heavy metal. According to the latest guidance provided by DOE's Office of Civilian Waste Management in their "Waste Acceptance System Requirements Document," Revision 3, April 1999, DOE spent nuclear fuel "may be accepted as bare fuel. The specific acceptance criteria for this bare fuel will be developed on a case by case basis." Therefore, the decision whether or not to treat spent nuclear fuel, including N-Reactor fuel, before emplacement in a geologic repository has not been made. As discussed in Section 1.2 of the EIS, the presence of metallic sodium is the primary but not the only reason for the proposed action. The presence of metallic uranium, or the presence of highly enriched uranium, could also complicate the process of certifying the repository if it accepted sodium-bonded spent nuclear fuel for disposal. Qualification of the spent fuel for disposal in a geologic repository would require sufficient data and predictive analyses to demonstrate that emplacement of the spent nuclear fuel would not adversely

Commentor No. 46: H. Zerriffi & L. Ledwidge (Cont'd)

DOE has not stated is that without this Action there would be no more funds available for EMT and DOE needs to make a decision now on whether to proceed with this technology. The "Purpose and Need for Action" should clearly state that these programmatic considerations, maintaining an EMT program and possibly using the Savannah River canyons which are slated to be shutdown, are driving the timing of this project. 46-12 (Cont'd)

In addition to the manner in which the purpose and need for this action have been presented, there are also serious problems with DOE's presentation of the health effects of radiation. Surprisingly, the DEIS indicates that the dose to the general population near Argonne National Laboratory – West is ten times higher under the "No Action Alternative" (which would not involve any processing) than for Electrometallurgical Treatment of the spent fuel. (2-48) When asked at the public meeting held in Crystal City, VA on August 31, 1999, the representative from the EIS team stated that this was an artifact of an assumption used for the draft EIS. It seems that the inventory of spent fuel was based on a 1995 Programmatic EIS rather than the actual inventory at the Idaho National Engineering and Environmental Laboratory. The final EIS will be revised to reflect actual conditions.[3] However, the draft EIS is the only opportunity for the public to comment on the health effects of this action and potentially decide whether or not to support processing this fuel. Presenting the data in this manner on such a crucial issue is misleading. 46-13

Perhaps even more egregious, however, is the presentation in the DEIS of radiation risk and the use of the linear no-threshold theory for calculating risk. The EIS states that "[C]alculations of health impacts based on the linear no-threshold theory may overstate the actual impacts of low radiation doses and should be viewed as an upper bound on the potential health effects." (page 4-6) This statement is dubious at best. It is contradicted by works produced by scientific bodies including the National Academy of Sciences' Committee on the Biological Effects of Ionizing Radiation (BEIR), the International Commission on Radiological Protection (ICRP), and, notably, the National Council on Radiation Protection and Measurements (NCRP): 46-14

- "[I]t must be presumed that even small radiation doses may produce some deleterious health effect." (ICRP, 1991. *1990 Recommendations of the International Commission on Radiological Protection, ICRP Publication 60*, paragraph 100, page 25)
- "[T]he probability of a cancer resulting from radiation increases with increments of dose, probably with no threshold." (ICRP, 1991. Paragraph S8, page 69)
- "In spite of evidence that the molecular lesions which give rise to somatic and genetic damage can be repaired to a considerable degree, the new data do not contradict the hypothesis, at least with respect to cancer induction and hereditary genetic effects, that the frequency of such effects increases with low-level radiation as a linear, nonthreshold function of the dose."-(BEIR V, 1990. *Health Effects of Exposure to Low Levels of Ionizing Radiation*, Committee on the Biological Effects of Ionizing Radiation, National Research Council, page 4)

[3] Our thanks go to Dr. Lyman of the Nuclear Control Institute for pointing out this fact and asking the question.

Response to Commentor No. 46 (Cont'd):

affect the repository's ability to protect the environment and worker and public health and safety. To ensure the requirements of the State of Idaho Settlement Agreement and Consent Order are met, and to facilitate disposal, DOE needs to reduce the uncertainties associated with qualifying sodium-bonded spent nuclear fuel for disposal. Appropriate treatment and management of the sodium-bonded spent nuclear fuel would significantly reduce complications related to disposal qualifications.

46-11: As described in Section 2.5.4 of the EIS, DOE evaluated PUREX processing as one of the alternatives for the treatment and management of sodium-bonded spent nuclear fuel. PUREX processing at SRS was included as a reasonable alternative in response to the National Research Council's recommendation that only PUREX processing would provide a viable alternative to the electrometallurgical treatment technology. However, since the sodium-bonded spent fuel contains metallic sodium, stainless steel, and zirconium, PUREX processing of this fuel would require the development and installation of a front-end process to ensure compatibility with the F-Canyon operation. Therefore, only the declad and cleaned blanket spent nuclear fuel, which is mainly depleted uranium metal and fission products, would be processed using PUREX at F-Canyon. In this process depleted uranium and plutonium metals would be separated from the fission products. The fission products would be vitrified as borosilicate glass in the Defense Waste Processing Facility, stored at the site, and transferred to a geologic repository. As explained in Section 4.1.2 of the EIS, the separated depleted uranium and plutonium would be stored at SRS pending a decision on their disposition.

46-12: DOE is committed to improving its environmental management practices; to operating its facilities in a manner that meets or exceeds all applicable environmental, safety, and health requirements; and to cleaning up its environmental problems. The focus of the EIS is to assess the potential environmental and health impacts associated with treatment and management of sodium-bonded spent nuclear fuel. Although not final, the latest guidance provided by DOE's Office of Civilian Waste Management in their "Waste Acceptance System Requirements Document," Revision 3, April 1999 (see Section 4.12.1 of the EIS) indicates it is highly probable that sodium-bonded spent nuclear fuel would not be acceptable in the geologic repository without some stabilization and/or removal of the metallic sodium. Stabilization of the spent nuclear fuel and/or removal of the metallic sodium will provide greater

- "The dose-dependent excess of mortality from all cancer other than leukemia, shows no departure from linearity in the range below 4 sievert (Sv), whereas the mortality data for leukemia are compatible with a linear-quadratic dose response relationship." (BEIR V, 1990. Page 5)

More egregiously, the DOE misrepresents the 1995 NCRP report it quotes in the EIS by taking the quote out of context[6]. Here is what the EIS does *not* include:

"3.5 Summary

"Taken as a whole, the body of evidence from both laboratory animals and human studies allows a presumption of a linear no threshold response at low doses and low-dose rates, for both mutations and carcinogenesis." (NCRP, 1995. *Principles and Application of Collective Dose in Radiation Protection, NCRP Report No. 121*, page 45)"

As more has been learned about the health effects of radiation, radiation risk estimates have been revised upwards. This indicates that, over time, exposure to radiation has been found to be more harmful than previously believed. For instance, the 1990 BEIR V report noted that, upon reassessment of A-bomb dosimetry, "lifetime risk of cancer attributable to a given dose of gamma radiation now appears somewhat larger than formerly estimated," (page 5) and that, "[t]he dose-dependent increase in the frequency of mental retardation in prenatally irradiated A-bomb survivors implies the possibility of higher risks to the embryo from low-level irradiation than have been suspected heretofore" (page 8). 46-14 (Cont'd)

The seventh BEIR committee (BEIR VII) has just convened to reassess the health effects to humans of exposure to low doses of ionizing radiation. There are many questions it faces before it can assess whether current risk estimates are too tight. As the attached September 3rd letter indicates, the BEIR VII committee has yet to consider the range of risks involved. It is very premature to claim that the linear no threshold hypothesis overstates the risks of low-level radiation, to say the least.

It is curious that this EIS, unlike some others, fails to include basic information about the health effects of radiation, and instead includes a weak dispute of the linear no threshold hypothesis. The DOE cannot credibly dismiss the linear no-threshold hypothesis, a long-held assumption in radiation and health circles, on the basis of one study and a misrepresentation of another, as it appears is the case in this EIS.

By failing to provide full and accurate information about what is and is not known about the health risks of radiation, the DOE, its EIS contractor SAIC, and the EIS itself lose credibility and public trust.

The section on health effects in this EIS should be rewritten to incorporate the aforementioned comments.

Not only is the DOE proposing an action which is clearly unnecessary at this time and entails programmatic risks since the final waste forms may not meet final repository 46-15

[6] On page 4-6, the EIS quotes the NCRP report from page 45: "...essentially no human data can be said to prove or even to provide direct support for the concept of collective dose with its implicit uncertainties on nonthreshold, linearity and dose-rate independence with respect to risk."

Section 1.6.3) and in planning the closure of its PUREX processing capabilities, DOE now needs to decide whether these processes are suitable for treating the remaining sodium-bonded spent nuclear fuel, or whether there is sufficient reason to delay a decision and wait for the development of other treatment technologies. Delaying the EIS could result in a loss of capability and of experienced, knowledgeable technical staff should DOE decide at a later date to use the electrometallurgical process to treat the sodium-bonded spent nuclear fuel. Section 1.2 of the EIS has been revised for clarification. DOE also has conducted four independent nonproliferation impacts assessments of the electrometallurgical treatment technology over the last 11 years. These assessments found the electrometallurgical treatment technology does not conflict with U.S. nuclear nonproliferation policy for this specific application, and have concluded that the electrometallurgical treatment technology is not capable of separating plutonium in a form that would be suitable for weapons production.

46-13: Air emissions under the No Action Alternative in the draft EIS were estimated using the adjusted values given in the No Action Alternative in the Programmatic Spent Nuclear Fuel EIS. The adjustment was based on the ratio of heavy mass inventory of the sodium-bonded spent nuclear fuel (60 metric tons) to the entire spent nuclear fuel inventory (274 metric tons) at INEEL. DOE assumed this estimate bounds any future degradation of the sodium-bonded spent nuclear fuel during storage at the INEEL site. The consequences resulting from this estimate were very small, and there was no intention to mislead the public. Since the issuance of the draft EIS, DOE has modified the activities under both options of the No Action Alternative as described in Section 4.2 of the final EIS, reevaluated the potential for sodium-bonded spent fuel degradation in wet and dry storage and revised the air emissions and associated health effects. The new results are provided in the final EIS.

46-14: As described in Section 4.1.3 of the EIS, the estimated health effects from radiation doses used in this EIS are based on the linear-no-threshold theory of radiation carcinogenesis. DOE would not consider any threshold in evaluating the potential cancer risk associated from radiation exposure, i.e., the limit of the range is extended to zero dose. As explained in Appendix E, Section E.2.2, of the EIS, there is a scientific uncertainty about cancer risk in the low-dose region below the range of epidemiologic observation, and the possibility of no risk cannot be excluded (from Committee on Interagency Radiation Research and Policy Coordination, Scienc Panel Report No. 9). DOE has revised the text in Section 4.1.3 of the EIS to remove the contentious

Commentor No. 46: H. Zerriffi & L. Ledwidge (Cont'd)

criteria, it is doing so in a way that poses significant nonproliferation problems. The Department should be commended for conducting a parallel Nonproliferation Assessment.[7] However, the resulting document seriously downplays some of the proliferation implications of the proposed actions, particularly Electrometallurgical Treatment (EMT).[8] The Nonproliferation Assessment does note many of the important proliferation risks posed by EMT:

- EMT can produce weapons-usable HEU
- EMT is a subset of a larger process which can separate plutonium and therefore has parallels with traditional reprocessing techniques such as PUREX
- EMT involves bulk processing which makes international safeguards harder to implement
- Safeguards have not been demonstrated since this a new technology.

However, the Assessment's system for grading the proliferation impacts, while a good start, was not sufficient. Each of the processing technologies was graded based on four policy factors[9] and three technical factors.[10] Each technology was graded on each factor. The grades were "fully meets nonproliferation objectives," "could raise nonproliferation concerns," and "raises nonproliferation concerns." Additionally, each of the DEIS alternatives (some of which involve using two different technologies for different portions of the fuel) were also graded on the same factors and with the same grading system. This system of grading poses a number of problems:

- There were no explicit criteria for choosing a particular grade for a particular technology or alternative.
- It was not made clear how mitigating factors were accounted for in the grading. There were a number of instances in which a proliferation concern was expressed, but a mitigating factor was also explained.

[7] United States Department of Energy, Office of Arms Control and Nonproliferation, *Nonproliferation Impacts Assessment for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel*. July 1999. (hereinafter referred to as "Assessment")

[8] This discussion of the nonproliferation impacts of actions proposed in this DEIS will focus on electrometallurgical treatment for three reasons. First, the original intent was to undertake an EIS on EMT. The scope was only expanded after the scoping hearings on the original EIS. Second, EMT would be used for five of the six alternatives presented (not including the "No Action Alternative.") Third, EMT has the potential for widespread global use under the rubric of either waste management or advanced nuclear fuel cycles (both as a stand-alone technology and as part of a more comprehensive pyro-processing system for which EMT is a crucial component). However, all of the technology choices have serious nonproliferation implications with the possible exception of repackaging in high-integrity cans of blanket spent fuel.

[9] Consistency with Nonproliferation policy, Avoiding encouragement of plutonium reprocessing, Building confidence that the United States is not producing materials for weapons, Supporting negotiations for a Fissile Materials Cut-off Treaty.

[10] Assuring against theft or diversion, Facilitating cost-effective International monitoring, Difficult-to-retrieve final form.

46-15 (Cont'd)

46-16

46-17

46-18

46-19

Response to Commentor No. 46 (Cont'd):

statement by providing a reference to the discussion provided in Appendix E, Section E.2.2.

46-15: See response to comment 46-3.

46-16: The commentor's support for conducting the Nonproliferation Impacts Assessment is noted. Although the assessment of nonproliferation impacts is not a part of the EIS process, none of the alternatives analyzed in this EIS, with the exception of PUREX processing at SRS, would generate weapons-usable fissile materials. Although highly enriched uranium would be an interim product, it would be down-blended to low-enriched uranium during electrometallurgical treatment.

46-17: Although assessment of nonproliferation impacts is not a part of the EIS process, it should be noted that electrometallurgical treatment is not capable of producing plutonium for nuclear explosive purposes. As conceived for the cancelled Integral Fast Reactor project, the liquid cadmium cathode would have produced a metal alloy product containing up to 70 percent plutonium, which could only have been obtained after subsequent processing in a high-temperature vacuum furnace. The balance of materials would be those elements most difficult to separate from plutonium by any chemical means, such as uranium, americium, neptunium, curium, and the rare earth fission products. The plutonium metal alloy product would have a high fission product and transuranic content, a high heat source, a high neutron radiation source, and a high gamma radiation source, any one of which would make the design of a weapon extremely difficult. Neutron and gamma radiation would be three to four orders of magnitude higher than weapons-grade or reactor-grade material. These levels of radiation are lethal and would prohibit any handling of the material or weapon by other than remote means. Development of the cathode progressed only to the point of technical feasibility. No prototype or working model was ever commissioned for the Fuel Conditioning Facility. During electrometallurgical treatment, plutonium would stay mixed with the fission products and electrolyte salt. The plutonium and fission products then would be immobilized in the ceramic waste form. The ceramic waste form is more resistant to plutonium recovery than the metallic waste forms that result under the other alternatives that employ melt and dilute technologies and high-integrity cans.

46-18: There are several features of the electrometallurgical treatment process that make it adaptable to international safeguards. The process cell, made

Commentor No. 46: H. Zerriffi & L. Ledwidge (Cont'd)

- Chapter Six, evaluating the individual technologies, only considered the US context (as clearly indicated by the title chapter).[11] This is insufficient as proliferation is clearly a global issue.
- It is not made explicit how the combinations of individual grades for different technologies were combined to derive an overall grade for each alternative.
- While EMT in the global context was discussed in terms of the four policy and three technical factors (Assessment, Section 3.5) there were no formal grades assigned. If a grading system was to be used, it should have been used consistently whenever a technology or alternative was being discussed in terms of the technical and policy factors. The result is that EMT was only graded in the U.S. context in Chapter Six.

46-19 (Cont'd)

The Assessment also downplays some of the proliferation risks, resulting in grades which are lower than they would be if a more comprehensive evaluation had been conducted. For example, there is little justification for the conclusion that EMT "*could* raise nonproliferation concerns" for facilitating cost-effective monitoring. This is a bulk processing method with fissile materials separations and *no demonstrated system for safeguards*. (Assessment, p. 6-3)

More puzzling is the conclusion that EMT fully maintains consistency with US nonproliferation policy. The DOE apparently reached this conclusion because there would be no separation of plutonium. However, this is a very narrow definition of US nonproliferation policy and objectives given the connection of EMT with systems that could result in plutonium separation. The Assessment itself notes in its section on the global implications of EMT that "both domestic and export applications of EMT pose concerns with respect to U.S. Nonproliferation policy." (Assessment, p. 3-15) The fact that the HEU would be diluted or that export controls would be put in place do not adequately mitigate the proliferation implications.

46-20

The DOE also concludes that EMT only "*could* raise concerns" about avoiding encouragement of plutonium processing. This is despite the statement that:

> Extending the time that U.S. separations facilities operate and using a separations process to prepare spent nuclear fuel for geologic disposal (while at the same time acknowledging that the fuel does not pose near-term safety and health vulnerabilities and that such processing technically is not required) could serve to undermine U.S. credibility in expressing concern to other countries about the proliferation problems associated with conventional reprocessing in the nuclear fuel cycle. (Assessment, p. 6-4)

46-21

The justification for the mid-level grade (rather than concluding that this would raise proliferation concerns) is that the US can make it clear that this is being done to address specific chemical requirements of a small batch of fuel, that plutonium is not separated, and that it is being done for spent fuel disposal "rather than as part of a breeder reactor fuel cycle." (Assessment, p. 6-4) This ignores the earlier statement (in the same paragraph) that the processing is not required in the near term. More significantly, it ignores the possibility that others would use the justification of spent fuel management to

[11] "Evaluation of the Technologies in the U.S. Context as Scoped in the *Draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel*."

Response to Commentor No. 46 (Cont'd):

moved in and out. These openings are secured and can be readily monitored for material transfers. There are no liquid waste streams through which materials can be piped out of the facility. All by-products and waste from the process would be in solid form, and so would be accountable by unit inventory. Finally, all by-products and waste moving out of the facility could be subjected to nondestructive examination if additional assurances were required under international safeguards agreements.

46-19: The assessment of nonproliferation impacts is not part of the scope of the EIS. However, the Nonproliferation Impacts Assessment for the management of DOE's sodium-bonded spent nuclear fuel was conducted to be consistent with nonproliferation assessments for other proposed DOE activities. A group of independent experts reviewed all the reasonable alternatives included in the draft EIS for nonproliferation considerations based on both policy and technology. While their conclusions are necessarily somewhat subjective, DOE is satisfied that the report represents a fair, unbiased view of the nonproliferation impacts of the alternatives. The report was reviewed and approved by the DOE Office of Arms Control and Nonproliferation prior to its issuance. DOE believes that the U.S. context is appropriate for the technical evaluation. The types of spent fuel that would be managed under the alternatives considered in the draft EIS are unique to U.S. research reactors. All activities would be carried out under the DOE safeguards and security requirements implemented to prevent the theft and diversion of nuclear materials, including spent fuel. The global implications have been considered under policy factors. The potential impacts of the various alternatives on U.S. nonproliferation policy are described in Chapter 6 of the Nonproliferation Impacts Assessment and in the conclusions of the assessment.

46-20: The United States' policy on nonproliferation is contained in Presidential Decision Directive 13, a classified document. At the time the Presidential Directive was signed, an unclassified press release stated that, "The U.S. will seek to eliminate where possible the accumulation of stockpiles of highly-enriched uranium or plutonium." This would be done by down-blending the highly enriched uranium in the driver spent nuclear fuel and immobilizing the plutonium in the ceramic waste form. The press release also stated that the United States "does not itself engage in plutonium reprocessing for either nuclear power or nuclear explosive purposes." Under the electrometallurgical treatment, the plutonium would be immobilized in the ceramic waste form

Commentor No. 46: H. Zerriffi & L. Ledwidge (Cont'd)

implement a full pyroprocessing system with plutonium separation. Research into such systems is already on-going in a number of countries and the U.S. implementation of a subset of pyroprocessing will clearly undermine U.S. credibility. 46-21 (Cont'd)

Electrometallurgical treatment should not be considered in isolation of its possible configuration for removal of plutonium nor without considering additional processing steps which could be implemented to further separate plutonium. While the current configuration does not result in separated plutonium, EMT should still be considered a reprocessing technology. Furthermore, EMT alone can result in separation of HEU and so should be considered a reprocessing technology on that basis. 46-20

As noted by the Nonproliferation Assessment, EMT can be modified in order to separate Pu, either by adding in a cadmium cathode and other equipment developed for pyroprocessing of IFR fuel or by processing the residual waste from EMT. What is not made explicit is the difference in proliferation barrier between spent fuel and EMT waste. The addition of the cadmium cathode to separate plutonium and a cathode processor would result in a product which is as high as 70% plutonium, 30% actinides and <1% rare earth fission products.[12] Aqueous processing to further separate the plutonium would be on a much smaller scale than that necessary for PUREX processing of the entire amount of spent fuel. The total amount of material to be processed would be about 100 to 1000 times less for the same amount of plutonium separation. Conceivably this could be done on a glove-box scale, assuming lower concern about worker health. The process would result in significantly less detectable air emissions since the volatile fission products would have already gone through the pyroprocessing stage and been emitted. The Nonproliferation Assessment does not discuss the effect this would have on implementation of international safeguards. 46-17

The Nonproliferation Assessment relies too heavily on the implementation of safeguards and on U.S. pronouncements as to the purpose of processing this particular spent fuel. These are inadequate mitigating factor. Safeguards do not have an absolute guarantee of success and are made more difficult by the types of processes discussed here. Generally, it would be best to avoid using these types of processing technologies, rather than rely on safeguards. 46-18

The Assessment also does not seem to integrate its discussion in Chapter three concerning the "Potential use of Electrometallurgical Treatment in a Global Context" with its evaluation about the U.S. context. The evaluation of EMT for this EIS cannot and should not be done with only the US context in mind. Both the US and global contexts need to be considered and the future of Electrometallurgical processing technologies needs to be integrated into the discussion. By separating these considerations, the Assessment seriously downplays the implications of EMT and reaches conclusions which cannot be supported when one looks at the overall picture. 46-19

In conclusion, there is no need for this action at this time. The spent fuel should be stored pending determination of Yucca Mountain's suitability as a repository and of 46-6 46-4

[12] U.S. Congress, Office of Technology Assessment, *Technical Options for the Advanced Liquid Metal Reactor – Background Paper*, OTA-BP-ENV-126 (Washington, DC: U.S. Government Printing Office, May 1994) p 21

Response to Commentor No. 46 (Cont'd):

46-21: As stated in the Nonproliferation Impacts Assessment, the alternatives involving PUREX reprocessing and broad application of electrometallurgical treatment of both driver and blanket fuel have a greater potential to provide encouragement to other countries to engage in plutonium reprocessing. Given the small quantity and unique characteristics of the sodium-bonded spent nuclear fuel and the reason for the treatment, however, such encouragement, if any, would be limited. Sodium-bonded spent nuclear fuel represents approximately 2 percent of DOE's spent nuclear fuel inventory.

46-22: The commentor's opinion that development of technologies such as GMODS and Plasma Arc processing on the bases that they do not involve fissile material separation, is noted. As discussed in Section 1.5 of the EIS, one of the decisions that DOE could make in the Record of Decision is to take no action now and promote the development of a less mature technology (like GMODS and Plasma Arc) or some other new treatment technology (see also Section 4.2 of the EIS).

Commentor No. 46: H. Zerriffi & L. Ledwidge (Cont'd)

Response to Commentor No. 46 :

final waste acceptance criteria for whatever high-level waste management strategy is finally chosen. Avoiding the complete close-out of the EMT project is not a sufficient reason for undertaking this project, both from a health and environment and a non-proliferation perspective. In the meantime, there may be justification in proceeding with technology development of alternatives to the proposed processes, such as GMODS and Plasma Arc processing, which do not have fissile material separation or the possibility of being configured for fissile material separation. 46-5 46-12 46-22

If the Department of Energy undertakes reprocessing of this spent fuel under the guise of environmental management it will set a dangerous precedent and significantly harm U.S. nonproliferation objectives. Electrometallurgical treatment, which appears to be the favored technology for at least part of the waste, poses particular concern because of its potential for clandestine plutonium separation, its potential use as a "waste management technique" in the context of "advanced" fuel cycles (with the explicit goal of plutonium separation) both in the United States and abroad and its use of a bulk process for which international safeguards have not yet been established. Pyroprocessing is an active area of inquiry in many nuclear countries, including the U.S., and continued U.S. interest in this technology and the application of a subset of pyroprocessing, would irreparably harm the U.S. government's credibility on nonproliferation. 46-15

Commentor No. 47: Ted L. Carpenter

SENT BY: 8-28-98 :11:11AM :SHOSHAN TRIBES-ATTY'S→ 3014261873:# 1/ 1

The SHOSHONE-BANNOCK TRIBES

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Dear Jeffery:

Thank-you for your assistance during the August 26 meeting in Idaho Falls. I appreciated the professional service provided by everyone involved in providing public information that evening

I have studied all of the material. My comments must not be considered "speaking for the Tribes." Only the Tribal Council officially does that. As the Tribal/DOE Project Environmentalist, I will comment.

Considering all of the material "All metric tons of heavy metal of sodium-bonded . . . fuel" to be "waste" for "disposal" seems worse than killing a Bison, taking the tongue (or the tongue and the hide), then leaving every other part of the animal to rot! Molybdenum, ruthenium, rhodium, palladium, uranium, zirconium, niobium, chromium, nickel, etc. are valuable resources that are expensive to mine and refine by environmentally responsible methods. 47-1

The electrometallurgical process seems to have the most potential for wise, environmentally responsible use of the Earth's resources. Your own documents (p. S 13 in the Summary) state that "the electrometallurgical treatment process uses electrorefining, an industrial technology used to produce pure metals from impure feedstock. Electrorefining has been used to purify metals for more than 100 years."

Alternatives 3 and 6 would produce increased environmental impact because of the thousands of miles of transportation required between the INEEL and the Savanna River Site. Fossil fuels would be burned to fuel the engines that transport these materials. Additionally, the transportation route crosses the Fort Hall Reservation of the Shoshone-Bannock Tribes. I encourage you to consider sodium-bonded driver fuel and blanket fuel treatment and management systems that minimize shipments of these materials across the reservation. Personally, I do not imagine any radiation hazard to the Reservation community because of these shipments, but the fears of many here are real. Because of those fears, I consider transportation across these Tribal lands to be a serious issue. 47-2

I have only one sort-of technical comment. On page S-2 of the Summary, I read, "Metallic sodium reacts vigorously with water or moist air, . . ." Obviously, sodium reacts vigorously with moist air because moist air contains water in the form of water vapor. Thus the "or" is incorrect. Water in a liquid or vapor form will react vigorously with sodium resulting in spontaneous combustion of the hydrogen gas that is released by the reaction. 47-3

Please feel free to contact me if you have any questions.

Sincerely,

Ted L. Carpenter, Ph.D.

Response to Commentor No. 47:

47-1: Most of the noble metal fission products (e.g., niobium, technetium, ruthenium, rubidium, silver, cadmium, and zirconium) and fuel alloy (zirconium) in the electrorefiners would remain with the fuel cladding hull in the anode basket. In addition, some actinides would also remain with the noble fission products. The amount of material retained in the anode basket would strictly depend on the electrorefining operation conditions. If more actinides and the fuel matrix were dissolved in the molten salts, the retention of noble fission products would be lowered. The metal remains in the anode basket would be radioactive, and would be classified as high-level radioactive waste. It is true that electrometallurgical treatment has been used to produce metals from impure feedstock. However, that impure feedstock included metals with chemical contamination, not radioactive isotopes of the same metals. Noble metal recovery from the metallic waste would have limited uses because the metal would still be radioactive (i.e., it would contain radioactive isotopes of the metal elements), and would still be considered radioactive metallic waste. However, uranium would be separated and could be used for other purposes. The disposition of this uranium, along with DOE's inventory of surplus uranium, will be determined through another NEPA review.

47-2: DOE assumes that the commentor is referring to Alternative 3 and 5 (not 6), in which the declad and cleaned (metallic sodium removed) blanket spent nuclear fuel would be transported to SRS for treatment. As explained in the EIS, the risks associated with the fuel transport are very small. Regardless of the alternative, DOE would need to transport spent nuclear fuel and/or high-level waste out of INEEL. DOE will proceed in accordance with the DOE/Shoshone-Bannock Tribes Agreement-in-Principal, which covers notification and coordination of the transport of radioactive materials across the Fort Hall Reservation. All risks, including transportation, are included in the EIS and will be considered by DOE prior to making any decisions regarding the treatment and management of sodium-bonded spent nuclear fuel.

47-3: The commentor is correct, metallic sodium reacts with water and, consequently, moist air. The text has been revised accordingly.

Commentor No. 48: Debra Patla

_____Forward Header_____
Subject: INEEL pyro
Author: dpatla@tetonvalley.net_at_INTERNET
Date: 10/4/99 11:14 PM

Dear Ms. Lesica,
I strongly object to "pyroprocessing" at INEEL. I was stunned to hear that plans are again being advanced for reprocessing spent nuclear fuel at the INEEL. I thought this bad idea was put to bed years ago when the Special Isotope Separator was finally rejected. **48-1**

I am an Idaho resident and a "down-winder". The DoE site at INEEL is viewed very negatively in this area (near the Teton Mountains), particularly since the plan was launched to incinerate nuclear wastes at INEEL. We don't trust the government to take care of our health and welfare. We fear that Idaho and Wyoming are a rural no-place to headquarters DoE, a throw-away zone because the population of humans is low, making INEEL a great place for creating and storing deadly elements. **48-2**

I am protesting not only as an Idaho resident. Weapons-grade plutonium and uranium is the last thing the world needs.

The EIS should reveal estimates of how much weapon-grade Plutonium and uranium exists, and how much may be in hands of elements hostile to the U.S. or other governments. It should estimate, given past trends and possible future scenarios, what is the likelihood that the products will be used to destroy life and/or induce global instability. **48-3**

The EIS should clearly document who initiated this project, and why. The trail of responsibility for this project should be made clear so that in the event of future disasters, the American and global public will know who is responsible.

The EIS should attempt to analyze how interrelationships of private industry and government officials contributed to the initiation of this project. It should assess the 'revolving door' whereby DoE and military personnel end up in related private industries after they leave government service. **48-4**

It should document what kinds of freedoms might end up being restricted if DoE institutes stricter security measures. **48-5**

The EIS should document how much waste will be generated by this project and where it will be stored, both temporarily and permanently. **48-6**

It should look at cumulative effects, and reveal how much waste exists at INEEL. It should analyze the costs of total clean-up, and reveal how much this project will add to total clean-up costs. **48-7** **48-8**

Response to Commentor No. 48:

48-1: The commentor's objection to electrometallurgical treatment (pyroprocessing) at INEEL is noted. This EIS evaluates several alternatives to electrometallurgical treatment including a No Action Alternative. The Special Isotope Separator referred to by the commentor was a weapons material production facility planned for INEEL back in the late 1980s. This facility was designed to use laser processing to produce weapons-grade plutonium from fuel-grade plutonium. The Special Isotope Separation Project EIS (DOE/EIS-0136) was published in November 1988. With the end of the Cold War, the need for plutonium production disappeared, and plans for the plutonium separation plant were halted. The special isotope separation laser process would not support the treatment and management and ultimate disposition of sodium-bonded spent nuclear fuel.

48-2: DOE has agreed to move all spent nuclear fuel out of the State of Idaho by 2035. To fulfill this commitment and prepare the fuel for ultimate disposal, DOE is proposing to treat and manage its sodium-bonded spent nuclear fuel at either INEEL or SRS.

48-3: Although the assessment of nonproliferation impacts is not a part of the EIS process, none of the alternatives analyzed in this EIS, with the exception of PUREX processing at SRS, would generate weapons-usable fissile materials. Although highly enriched uranium is an interim product, it is downblended to low-enriched uranium during electrometallurgical treatment. Alternative 3, PUREX processing, is the only alternative that would generate weapons-usable fissile material, including plutonium. This plutonium would be managed along with other surplus plutonium as described in the Surplus Plutonium Disposition EIS. The SBSNF EIS has been prepared in accordance with NEPA, Council on Environmental Quality regulations on implementing NEPA (40 CFR 1500-1508), and DOE's NEPA implementation procedures (10 CFR 1021). As discussed in the introduction, the basic objective of this EIS is to provide the public and DOE decision-makers with a description of the reasonable alternatives for the treatment and management of DOE's sodium-bonded spent nuclear fuel and their potential environmental impact. Estimating how much plutonium and uranium exists and the likelihood of these materials being used to destroy life and/or induce global instability are beyond the scope of the EIS.

Commentor No. 48: Debra Patla (Cont'd)

The EIS should attempt to measure how much risk to U.S. citizens this project poses, both in the present, 20 years from now, and long into the future. This analysis should be reviewed by impartial sources rather than DoE. Their commentary should be included within the EIS. 48-9 48-10

I urge DoE to start putting more money into environmental clean-up and protection rather than producing elements of potential mass murder. 48-11

Sincerely,
Debra Patla
PO Box 230
Victor, ID 83455

Response to Commentor No. 48 (Cont'd):

48-4: The Electrometallurgical Treatment Research and Demonstration Project was initiated by DOE with Congressional funding to demonstrate electrometallurgical treatment technology, as directed by the 1995 Record of Decision for the Spent Nuclear Fuel Programmatic EIS (60 FR 28680). Near completion of the demonstration project, DOE developed this EIS to evaluate the potential environmental impacts of using electrometallurgical treatment or other technologies to treat the remaining sodium-bonded spent nuclear fuel and reduce the risk that the sodium-bonded spent nuclear fuel would not be accepted in a geologic repository. Chapter 1 of the EIS discusses the purpose and need for the proposed action. All preparers of the EIS, their organization, responsibilities, education, experience, and technical expertise are listed in Chapter 7 of the EIS. Council on Environmental Quality regulations 40 CFR 1506.5(c), which have been adopted by DOE (10 CFR 1021), require contractors preparing this EIS to execute a disclosure statement specifying they have no financial or other interest in the outcome of the project. This disclosure statement is provided in Appendix L of the EIS. Analyzing private industry and government interrelationships and the actions of DOE and military personnel after they leave government service are beyond the scope of this EIS.

48-5: The proposed action of the EIS does not require any changes in security.

48-6: The amount and form of the waste generated under each alternative are discussed in Chapter 4 of the EIS. The discussions in the chapter identify the final disposition of each waste form produced. For example, as described in Section 4.3.6, the ceramic and metallic high-level radioactive waste generated under Alternative 1 (electrometallurgically treat blanket and driver fuel at ANL-W) would be temporarily stored at the Radioactive Scrap and Waste Facility, and when a geologic repository is available the waste forms would be removed from storage and transferred to INEEL's Dry Transfer Facility for packaging and shipment to the repository.

48-7: Section 4.11.1.6 of the EIS summarizes cumulative waste generation at the INEEL site. This includes all waste currently present at the site, plus any new waste to be generated in the reasonably foreseeable future.

48-8: The SBSNF EIS has been prepared in accordance with NEPA, the Council on Environmental Quality regulations on implementing NEPA (40 CFR 1500-1508), and DOE's NEPA implementation procedures

Commentor No. 48: Debra Patla

Response to Commentor No. 48 (Cont'd):

the public and DOE decision-makers with a description of the reasonable alternatives for the treatment and management of DOE's sodium-bonded spent nuclear fuel and their potential environmental impact. However, DOE has issued a separate Cost Study that analyzes and compares the cost of the alternatives analyzed in the EIS. Cost will be considered during the decision-making process in preparing the Record of Decision.

48-9: DOE proposes to use the electrometallurgical treatment process to treat the sodium-bonded spent nuclear fuel and facilitate its ultimate disposal in a geologic repository. This process would transform about 60 metric tons of heavy metal sodium-bonded spent nuclear fuel into two inherently stable solid high-level waste forms. The process would take about 13 years to complete. Section 4.2 of the EIS discusses current risks to the public residing within 80 kilometers (50 miles) of the facilities where the sodium-bonded nuclear spent fuel is currently stored. The risks from operation of the electrometallurgical treatment process to the projected population (assumed to exist in the year 2010) residing within 80 kilometers (50 miles) of the facility are provided in Section 4.3 of the EIS. As explained in this section, the maximum annual dose to an individual from operation of this process is estimated to be less than 0.0004 millirem, or about 0.0001 percent of the background radiation dose. As explained in Section 4.3.6, the solid high-level waste would be packaged in special canisters and stored temporarily at the site. While in storage, this waste form would not pose any risks to any member of the public. This waste form is expected to be transferred to a geologic repository by 2035. The long-term impact from storage of this waste is evaluated in the Yucca Mountain EIS, which was issued in July 1999.

48-10: While the EIS has undergone internal DOE review, the NEPA public participation process provided an opportunity for all interested parties, including members of the public and Federal, state, local, and tribal officials, to independently review and comment on the draft EIS. All comments, along with DOE's responses, are included in the this final EIS.

48-11: Congress determines how funds are allocated. DOE spends monies consistent with Congressional direction. DOE is not in a position to make the difficult tradeoffs that may be required between alternative Federal programs and spending priorities. The issue of whether to fund the treatment and management of sodium-bonded spent nuclear fuel is beyond the scope of the SBSNF EIS. However, implementation of any of the sodium-bonded spent nuclear fuel treatment and management alternatives would not take

Commentor No. 48: Debra Patla

Response to Commentor No. 48 (Cont'd):

taxpayer dollars away from other environmental cleanup projects at INEEL. Each year Congress appropriates funds for environmental cleanup projects which are administered by the DOE Office of Environmental Management. The INEEL environmental cleanup efforts receive most of their money from these funds. Congress appropriates separate funds for spent nuclear fuel treatment, and these funds are administered by the DOE Office of Nuclear Energy, Science and Technology. The two sources of funds do not compete with each other.

Commentor No. 49: Kathleen E. Trever

INEEL Oversight
Idaho's eyes on the INEEL

Dirk Kempthorne, Governor
Kathleen E. Trever, Coordinator

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September 28, 1999

Susan M. Lesica, EIS Document Manager
Office of Nuclear Facilities Management
Office of Nuclear Energy, Science and Technology
U.S. DOE, NE-40
19901 Germantown Road
Germantown, MD 20874-1290

Re: DEIS for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel

Dear Ms. Lesica:

The State of Idaho INEEL Oversight Program submits the following comments on the Draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel. Section and page references precede specific comments.

COMMENTS

Summary, page S-9, 1st paragraph, line 3; Volume 1, Section 1.6.1.7, pages 1-9 and 1-10
"The proposed Advanced Mixed Waste Treatment Facility would treat...." The Record of Decision for the AMWTP EIS was issued in March 1999. Therefore, "proposed" should be "planned", and "would" should be "will".

Summary, page S-9, 4th item in italics; Volume 1, Section 1.6.2.3, page 1-11
The correct name of the referenced EIS is *Idaho High-Level Waste and Facilities Disposition Environmental Impact Statement.* 49-1

Summary, page S-10, last paragraph; Section S.8 Glossary; Volume 1, Section 2.2.1, page 2-3; Chapter 6 Glossary; Volume 2, Section D-2, page D-2
The only definition of fissium appears on page S-10; the term is used without definition in Volume 1, page 2-3 and in Volume 2, page D-2 (and perhaps elsewhere). "Fissium" should be added to the glossaries in Section S.8 and Chapter 6.

Summary, page S-13 to S-17; Volume 1, Sections 1.1 and 2.4
In each of these sections, the treatment methods that have been considered and eliminated should be identified as such and discussed in a separate section. It is confusing to read descriptions of GMODS, Chloride Volatility, etc. only to find out later that they have been eliminated from consideration. 49-2

an Idaho state program that independently monitors activities at the INEEL on behalf of the citizens of Idaho
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Response to Commentor No. 49:

49-1: The text cited by the commentor has been revised as appropriate. The name of the referenced EIS has been corrected. The term "fissium" has been added to the glossaries in Section S.10 and Chapter 6 of the EIS. The language used to explain or define "fissium" in Section S.2.1 of the Summary is also used in Section 2.2.1 of the draft EIS.

49-2: The purpose of Sections 1.1 and 2.4 of the draft EIS, as well as corresponding sections in the Summary, is to inform the reader of the pertinent characteristics of all potential technologies considered prior to selection of the reasonable alternatives presented in Section 2.6. Reasons why some of the technologies were dismissed from consideration as reasonable alternatives are found in Section 2.7. To avoid the confusion mentioned by the commentor, Section 2.3 of the final EIS has been revised to identify the dismissed technologies at an earlier point in the EIS.

49-3: Discharge waters to ANL-W's Industrial Waste Pond or Sanitary Sewage Lagoons are not waters of the U.S. and are exempted from compliance under the NPDES. However, these waters are designated as waters of the State of Idaho and, as such, require compliance with the state regulations that govern application of nonhazardous liquid waste (i.e., Land Application Permits). ANL-W applied to the State of Idaho for Land Application Permits for the Industrial Waste Pond and Ditches and the Sanitary Waste Treatment Pond Land Application Area on March 15, 1996, and July 17, 1998, respectively. ANL-W routinely monitors the effluent discharges to make sure they are within the limits identified in the Land Application Permits. The text of the various EIS sections of concern was revised to clarify that discharges are regulated in accordance with Idaho Land Application Permit requirements.

Commentor No. 49: Kathleen E. Trever (Cont'd)

Summary, Page S-31, 2nd paragraph; page S-37, 7th paragraph; Vol 1, Section 4.2.2, page 4-8, 3rd paragraph; Vol 1, Section 4.3.2, page 4-18, 3rd paragraph; Vol 1, Section 4.4.2, page 4-27, 2nd paragraph; Vol 1 Section 4.6.2 page 4-52, 5th paragraph; Vol 1, Section 4.8.2, page 4-76, 4th paragraph.

The statement "...discharges of nonhazardous liquid waste, which are monitored and subject to National Pollutant Discharge Elimination System (NPDES) permit requirements" implies that there is an NPDES permit for the industrial waste pond at ANL-W. However, the NPDES database does not include a permit for ANL-W. Further, the ANL-W Environmental Surveillance Report for 1997 did not refer to an NPDES permit. If there is an NPDES permit, its number should be included in section 3.2.4.1. References should also be identified to support the statement that the liquid waste is "nonhazardous" (e.g., results of TCLP analyses). **49-3**

Volume 1, Section 1.6.1.6, page 1-9; Section 3.2.11.9, pages 3-30 and 3-31

This section should note that DOE issued a Record of Decision on high-level waste from the Programmatic Waste Management EIS in August 1999. **49-4**

Volume 1, Section 1.6.2.2, page 1-11; Volume 1, Section 4.11.2, pages 4-103 to 4-105

These sections should reference the "Draft Environmental Impact Statement for a Geologic Repository..." issued in July 1999. **49-5**

Volume 1, Section 1.6.2.3, page 1-11, 1st paragraph under 1.6.2.3

"This EIS evaluates treatment alternatives for wastes that actions proposed in the SBSNF EIS could generate." The State of Idaho is familiar with the waste treatment alternatives in the forthcoming high-level waste EIS, but is unaware of the wastes referred to in this statement. The SBSNF EIS should identify what material and what quantities are or will be included. **49-6**

Volume 1, Section 2.4.1, Figure 2-2, page 2-7

The lower horizontal line between "Metal Casting" and "Cathode Processing" in the figure has arrowheads on both ends; it should have only one, probably on the left. **49-7**

Volume 1, Section 2.4.4, pages 2-10 and 2-11; Sections 2.6.4, 2.6.5, 2.6.6, and 2.6.7, pages 2-35 to 2-38; Sections 4.6, 4.7, and 4.8; Volume 2, Section C.4

What is the waste classification of the "metal waste form" or "melt and dilute product" that would be produced under the melt and dilute alternatives? Also, these alternatives only describe storage pending disposal; where would this material be disposed? Do these alternatives create a waste form whose ultimate disposal would be problematic? Would it be acceptable at the geologic repository? Is its isotopic content included in Appendix A of the draft EIS for the Geologic Repository? **49-8**

Volume 1, Section 2.6.4, pages 2-34 and 2-35; Section 4.1.2, page 4-3; Section 4.5.6, pages 4-47 to 4-51 **49-9**

Response to Commentor No. 49 (Cont'd):

49-4: The text cited by the commentor has been revised to incorporate the latest Record of Decision for DOE's Waste Management Program: Storage of High-Level Radioactive Waste (64 FR 46661). In this third decision, DOE would store immobilized high-level radioactive waste in a final form at the site of generation (Hanford, INEEL, SRS, or the West Valley Demonstration Project in New York) until transfer to a geologic repository.

49-5: The text cited by the commentor has been revised to state that the Yucca Mountain Draft EIS was published in July 1999. The equivalent section in the Summary was also revised to reflect this change of status in the ongoing NEPA actions.

49-6: The sentence identified by the commentor in the SBSNF Draft EIS is no longer correct. At the time this sentence was written, it was unclear what role, if any, the Idaho High-Level Waste and Facilities Disposition EIS would play in the treatment of waste generated by the treatment of sodium-bonded spent nuclear fuel at ANL-W. Since that time, it has been determined that the high-level radioactive waste generated by the treatment and management of sodium-bonded spent nuclear fuel at ANL-W would not require any additional treatment at INEEL and are not within the scope of the Idaho High-Level Waste and Facilities Disposition EIS, which only evaluates the treatment of specific amounts of calcined high-level and sodium-bearing, radioactive waste material currently located at INEEL. Section 1.6.2.3 has been revised.

49-7: The commentor is correct. The figure has been revised.

49-8: For the purposes of this EIS, the "metallic waste form" or "melt and dilute product" from the melt and dilute alternatives are considered to be high-level radioactive waste that would be disposed of in a geologic repository. Disposal of the metallic waste form or melt and dilute product from the melt and dilute alternatives in the geologic repository is not expected to be problematic. The Yucca Mountain Draft EIS assumes that all sodium-bonded spent nuclear fuel will be treated using the electrometallurgical process (Alternative 1 of this EIS) and the Yucca Mountain Draft EIS presents isotopic contents in its Appendix A that are in accordance with the electrometallurgical treatment process.

49-9: The amount of plutonium in the various sodium-bonded spent nuclear fuel is given in Appendix D, Section D.2. Section 4.1.2 of the EIS has been modified to provide a perspective on the amount of plutonium that would be separated

Commentor No. 49: Kathleen E. Trever (Cont'd)

One or more of these sections (as well as the appendix) should include the amount of plutonium that would be produced and compare it to the total amount of plutonium that is expected to be produced and stored at SRS. A summary of quantities would be useful for reviewers. **49-9 (Cont'd)**

Volume 1, Section 3.2.4, Figure 3-3, page 3-11
The reference "LMITC 1977" at the bottom of the figure should probably be "LMITCo 1997." **49-10**

Volume 1, Section 3.2.4.1, page 3-9, 5th paragraph
The text states "No flood maps of the Big Lost River are available...." This statement is not entirely true as maps of the flood plain are presented in Berenbrock and Kjelstrom (1998). Further, the reference given for the statement regarding the lack of flood plain maps (i.e. Abbott, Crockett, and Moor, 1997) is a predecisional draft. This EIS should cite the *original scientific study*. **49-11**

Volume 1, Section 3.2.4.1, page 3-9, 5th paragraph
The text states that "Flood diversion facilities... secured the INEEL from the 300-year flood" and references an EIS (i.e. DOE, 1996c) as support for this statement. Instead of citing another EIS, this EIS should cite the *original scientific study* that reached that conclusion. **49-12**

Volume 1, Section 3.2.4.1, page 3-9, Last paragraph
Provide a reference for the statement that the liquid waste is "nonhazardous." **49-3**

Volume 1, Section 3.2.4.2, page 3-12, 2nd paragraph
Several of the statements in this paragraph should be referenced. For example, the text should cite the Federal Register for the sole source aquifer designation for the Snake River Plain aquifer. Scientific studies (not another EIS) should be cited for the amount of water in the aquifer and the source of recharge. **49-13**

Volume 1, Section 3.2.4.2, page 3-12, 3rd paragraph
Perched water bodies at the INEEL are present over relatively small areas of the site that are near surface water bodies or other sources of recharge. If perched water does not occur near ANL-W, then the last two sentences are irrelevant and should be deleted. The statement that "perched water tables tend to slow the migration of pollutants" should reference the original scientific study and not merely reference another EIS. **49-14**

Volume 1, Section 3.2.4.2, page 3-12, 4th paragraph
The EIS should provide a reference for the statement that the "tritium... concentration dropped 93 percent between 1961 and 1994." Also, the contaminant list provided is not all- inclusive. This section should also reference some of the more recent USGS **49-15**

Response to Commentor No. 49 (Cont'd):

processing compared to the total amount of plutonium (considered surplus plutonium) currently stored at SRS.

49-10: The reference cited by the commentor has been revised. The reference is now DOE 1999a, "Idaho National Engineering and Environmental Laboratory Advanced Mixed Waste Treatment Project Final Environmental Impact Statement," DOE/EIS-290, Office of Environmental Management, Idaho Operations Office, Idaho Falls, Idaho.

49-11: The EIS text was revised to more clearly indicate the availability of the preliminary study of the 100-year peak flow of the Big Lost River. The sentence containing the Abbott, Crockett, and Moor (1997) reference has been deleted. The EIS cites the original scientific study written by Berenbrock and Kjelstrom (i.e., USGS 1998).

49-12: DOE based the affected environment discussions on the Surplus Plutonium Disposition EIS, except where otherwise noted. The discussion of flood diversion facilities is provided in that document, which is readily available to the public, so no additional reference is necessary. It is accepted practice for DOE to cite peer-reviewed, published, and approved DOE documents.

49-13: DOE based the affected environment discussions on the Surplus Plutonium Disposition EIS, except where otherwise noted. The discussion of the Snake River Plain aquifer is provided in that document, which is readily available to the public so no additional reference is necessary. It is an accepted practice for DOE to cite peer-reviewed, published, and approved DOE documents.

49-14: DOE based the affected environment discussions on the Surplus Plutonium Disposition EIS, except where otherwise noted. The discussion of the Snake River Plain aquifer is provided in that document, which is readily available to the public so no additional reference is necessary. It is accepted practice for DOE to cite peer-reviewed, published, and approved DOE documents.

49-15: DOE based the affected environment discussions on the Surplus Plutonium Disposition EIS, except where otherwise noted. A discussion of historical tritium concentrations is provided in that document, which is readily available to the public so no additional reference is necessary. Text in Section 3.2.4.2 was revised to address the migration of waste into the aquifer. The list of groundwater contaminants is intended to show examples of known contaminants and indicate those of primary concern. Text has been added to the this EIS to refer the reader to the annual environmental reports for more information on groundwater monitoring programs.

Commentor No. 49: Kathleen E. Trever (Cont'd)

hydrologic conditions reports for the INEEL (e.g., Bartholomay and others, 1997) for readers who would like more information.

The statement "Components of nonradioactive waste entered the aquifer as a result of past waste management practices" uses the past tense, and consequently is misleading. Nonradioactive wastes continue to migrate to the aquifer at the INEEL. For example, chloride discharged to the percolation ponds at the INTEC migrates rapidly enough that chloride concentrations in groundwater mirror the concentrations in the wastewater (see Bartholomay and others, 1997; p. 36). **49-15 (Cont'd)**

Volume 1, Section 3.2.5, page 3-13, 1st paragraph
References should be provided for the age of the rhyolitic rocks. Also, the last two sentences appear contradictory. Further discussion of sinkhole and lava tube issues is appropriate. **49-16**

Volume 1, Section 3.2.5, page 3-13, 3rd paragraph
The reference given for the statement regarding capable faults (i.e. Abbott, Crockett, and Moor, 1997) is a predecisional draft, which hardly seems appropriate. Reference the *original scientific study*. **49-17**

Volume 1, Section 3.2.5, page 3-13, 4th paragraph
The statement "No earthquakes have been recorded within 48 kilometers of the site" is false. In fact, several small earthquakes have been recorded beneath the INEEL (Jackson and others, 1993). **49-18**

Volume 1, Section 3.2.5, page 3-13, 4th paragraph
"An earthquake with a maximum horizontal acceleration of 0.15 g is calculated to have an annual probability of occurrence of 1 in 5,000 at a central INEEL location." The EIS should list the reference for this important value. **49-19**

Volume 1, Section 3.2.5, page 3-13, 5th paragraph
The statement "most of the basaltic volcanic activity occurred at the Craters of the Moon National Monument 20 kilometers southwest of INEEL between 4 million and 2,100 years ago" is inaccurate. While the most recent volcanism on the Snake River Plain occurred at the Craters of the Moon National Monument, researchers have mapped five volcanic zones on the INEEL. ANL-W lies within one of these, the axial volcanic zone (Hackett and Smith, 1994). Most or all of the basalt flows on the INEEL were derived from these or other local eruptive centers. The text should be revised to discuss the volcanic activity on the INEEL. **49-20**

The text states "The probability of volcanic activity affecting facilities at the INEEL is very low." "Very low" is not quantitative, and thus has no real meaning. A quantitative estimate for volcanism at ANL-W can and should be derived using the data presented

Response to Commentor No. 49 (Cont'd):

49-16: The reference for the age of the rhyolitic rocks has been added to the EIS, and the two sentences referenced by the commentor have been modified for clarity.

49-17: Although Abbott, Crockett, and Moor (1997) is a predecisional draft, neither a draft or final version of the document will be issued. However, the document will be included in the Administrative Record for the EIS and will, therefore, be available to the public.

49-18: The statement that no earthquakes have been recorded within 48 kilometers (30 miles) of INEEL has been deleted from the EIS and reference to the occurrence of several "microearthquakes" at the site has been added (per Jackson et al. 1993).

49-19: The following reference has been added to the end of the sentence in question. Barghusen, J., and R. Feit, 1995, Technical Report on Affected Environment or the DOE Sites Considered in the DOE Waste Management Programmatic Environmental Impact Statement, META/Berger-SR-01, META/Berger, Gaithersburg, MD, July."

49-20: The referenced paragraph in Section 3.2.5 of the EIS has been revised using Hackett and Smith (1994). Also, reference to the volcanic zone within which ANL-W occurs has been added to the last paragraph of Section 3.2.5 of the EIS.

49-21: Although Abbott, Crockett, and Moor (1997) is a predecisional draft, neither a draft or final version of the document will be issued. However, the document will be included in the Administrative Record for the EIS and will, therefore, be available to the public.

49-22: Information presented in the second through fourth sentences of the referenced paragraph in Section 3.2.5 of the EIS is from ANL 1999a. This reference is provided at the end of the fourth sentence. The last sentence concerning disturbed soils has been retained.

49-23: The socioeconomic region of influence is not determined by proximity, but is defined by the areas where INEEL employees and their families reside, spend their income, and use their benefits, thereby affecting the economic conditions of the region. The region of economic influence was determined to be a four-county area in Idaho (Bonneville, Bingham, Bannock, and Jefferson Counties) in which large populations (94.4 percent) of all INEEL employees reside. The seven-county area used in other INEEL EISs was

Commentor No. 49: Kathleen E. Trever (Cont'd)

In the Hackett and Smith study (1994). The text should present this information, not merely reference another EIS. The last sentence should be deleted and replaced with a detailed discussion in this document. **49-20 (Cont'd)**

Volume 1, Section 3.2.5, page 3-13, last paragraph
The reference given for the statement regarding capable faults (i.e., Abbott, Crockett, and Moor, 1997) is a predecisional draft. The EIS should reference the *original scientific study.* **49-21**

The EIS should provide a reference or topographic map to support the statement that ANL-W lies within a closed basin. **49-22**

The statement "Soils are highly disturbed within developed areas of the site" is intuitive and could be deleted.

Volume 1, Section 3.2.8, page 3-18
The region of influence should be expanded. A seven-county area has been used for other INEEL EISs. Butte County should at least be included, since most of the INEEL site is in Butte County. **49-23**

Volume 1, Section 3.2.11.1, Table 3-10, page 3-25
Because DOE considers the sodium-bearing waste in the INTEC Tank Farm mixed transuranic waste that may be processed in the future at the NWCF, the box opposite "INTEC NWCF" under "Mixed TRU" should contain an "x." Notably, the State of Idaho considers this material to be high-level waste. **49-24**

Volume 1, Section 3.2.11.2, page 3-27, lines 2-4
"Most aqueous solutions...were concentrated by evaporation and separated into low-level radioactive waste streams..." should be changed to say "...separated into low-level and high-level radioactive waste streams...." **49-25**

Volume 1, Section 3.2.11.2, page 3-7, lines 4-11
"This calcination was completed in February 1998." Calcination of non-sodium bearing HLW has been completed, but calcination of sodium-bearing waste has not.

The sentences at the end of this paragraph could be interpreted as indicating that storage tanks are empty. These sentences should be clarified, since about 1.4 million gallons of liquid mixed sodium-bearing waste remain in the INTEC Tank Farm. **49-26**

Volume 1, Section 4.1.2, page 4-3 (and various subsequent discussions of TRU waste)
"Transuranic waste....This waste could be disposed of in the Waste Isolation Pilot Plant." Would the TRU waste resulting from this process be acceptable for disposal **49-27**

Response to Commentor No. 49 (Cont'd):

49-24: DOE concurs with the commentor, and this table has been revised in the EIS to reflect the change.

49-25: DOE has revised Section 3.2.11.2 of the EIS to be consistent with the information given in the Idaho High-Level Waste and Facilities Disposition Draft EIS.

49-26: See response to comment 49-25.

49-27: All of the transuranic waste generated by the treatment of sodium-bonded spent nuclear fuel would be acceptable for disposal at the Waste Isolation Pilot Plant under current regulations. If necessary, the Advanced Mixed Waste Treatment Facility will treat the waste to meet the Waste Isolation Pilot Plant Waste Acceptance Criteria and applicable requirements of the Toxic Substances Control Act and Resource Conservation and Recovery Act (RCRA) Land Disposal Restrictions.

49-28: For the purposes of evaluation, this EIS assumes that high-integrity can packaging could start as early as 2003. DOE would not begin packaging in high-integrity cans until it receives some indication that high-integrity can packaging would be acceptable under the waste acceptance criteria for the geologic repository and a high-integrity can specification is in place.

49-29: As described in the EIS, the adsorbent used in the off-gas system to collect volatile radionuclides released from spent nuclear fuel when it is heated is considered a high-level radioactive waste. This adsorbent material would be packaged and disposed of similar to other high-level radioactive waste generated under the proposed action. This high-level radioactive waste would be generated at ANL-W (and/or SRS), and would be stored and disposed of in a similar manner to the ceramic and/or metallic waste.

49-30: The text in Table 4-64 was revised to reflect this new information. The information presented in this table, as referenced, came from the Advanced Mixed Waste Treatment EIS released by DOE Idaho Operations in January 1999. DOE recognizes that there will always be other new commercial businesses that contribute to the cumulative impacts in the region. Since the potential incremental effects from the proposed action on the region would be small, it is not necessary to identify each of these new commercial businesses. As explained in Section 4.11.1 of the SBSNF EIS, DOE recognizes there are a number of existing and planned industrial and commercial facilities located in the counties surrounding INEEL, although the EIS does not identify them by name. Because of the distances between

Commentor No. 49: Kathleen E. Trever (Cont'd)

at WIPP under the WIPP Land Withdrawal Act and current regulations, or would legal changes be required for disposal at WIPP? | 49-27 (Cont'd)

Volume i, Section 4.4, page 4-26
Will geologic repository waste acceptance criteria and specifications for high-integrity cans be established by 2003? This information will be necessary for placement in cans to begin. | 49-28

Volume 1, Section 4.4.6, page 4-33, under the header "Other Associated Process High-Level Radioactive Waste", 1st paragraph; Section 4.5.6, page 4-48; Section 4.6.6, page 4-59; Section 4.7.6, page 4-72; Section 4.8.6, pages 4-82 and 4-83
"...absorbant used in the off-gas system which has collected the volatile radionuclides...." How and where will this material be treated and disposed? | 49-29

Volume 1, Section 4.10.1, Table 4-62, page 4-88
The Target store listed in this table is not a "reasonably foreseeable offsite action." It has been completed and open for business for some time. It is unclear why this store was included in this table and other similar projects were not. For example, other commercial businesses of similar size recently have been or are being constructed in the Idaho Falls area. | 49-30

Volume 1, Section 4.11, page 4-102
"The programmatic considerations presented below is a programmative perspective of the alternatives vis-a-vis the current regulatory environment...." This sentence should be rewritten to be a clear statement. | 49-31

Volume 1, Section 5.1.3, Table 5-1, pages 5-6 and 5-7
DOE Order 435.1 and the associated manual and guidance document are now final. DOE Order 435.1 should be included in the table and its implications fully discussed in the text. Also, the entire SBSNF EIS should be reviewed for consistency with Order 435.1 requirements for radioactive waste management and waste type definitions and modified as necessary. | 49-32

Volume 2, Appendix A, Table A-3, page A-17, 1st item under "Transportation"
"It is DOE's intention to minimize transport of radioactive materials associated with its sodium-bonded spent nuclear fuel inventory wherever possible." This statement should be qualified in the text, since the SBSNF EIS considers alternatives that do not minimize transport. | 49-33

Volume 2, Appendix A, Table A-3, page A-21, 3rd item
Is the sale of low-enriched uranium as feedstock for commercial reactor fuel consistent with the U.S. policy on reprocessing? | 49-34

Response to Commentor No. 49 (Cont'd):

INEEL and these facilities, there is no opportunity for interaction and no measurable contribution to the cumulative impacts.

49-31: The text cited by the commentor has been revised.

49-32: DOE Order 435.1 has been added to Table 5-1 of the EIS. This DOE Order replaces DOE Order 5820.2A, which was removed from the table. The definitions of radioactive waste materials identified in the EIS are consistent with the definitions used in DOE Order 435.1. The implications of DOE 435.1 are discussed, as appropriate, throughout the EIS.

49-33: DOE considered two alternative locations for the treatment and management of sodium-bonded spent nuclear fuel, INEEL and SRS. SRS was selected in response to the National Research Council's recommendation that only PUREX processing would provide a viable alternative to the electrometallurgical treatment technology. This is consistent with the statement made in Section A.1.3 of Appendix A that DOE would minimize transportation activities "wherever possible." As described in Section 4.9 of the EIS, the environmental impacts of transporting spent fuel to SRS are very small, and are essentially indistinguishable from those associated with local transport at the INEEL site.

49-34: Disposition of DOE's inventory of surplus uranium is not within the scope of this EIS. However, it will be the subject of a future NEPA action.

49-35: The definition of mixed waste in presented in Section B.5.1 of Appendix B has been expanded to indicate that mixed waste could be any radioactive waste that includes hazardous components, i.e., it could be either high-level radioactive, low-level radioactive, or transuranic waste.

49-36: The designation "Other Waste" has been removed from the list of waste types.

49-37: As part of the PUREX processing of spent nuclear fuel, the separated, impure plutonium would go through various cleaning cycles to reduce transuranic contamination. The separated plutonium from the blanket spent nuclear fuel would be considered surplus plutonium.

49-38: Qualifying statements were added to the table to clarify the radiation exposure units.

49-39: The text cited by the commentor has been revised.

49-40: DOE agrees with the commentor. The text has been revised for clarity and

Commentor No. 49: Kathleen E. Trever (Cont'd)

Volume 2, Section B.5.1, page B-7
The definition of mixed waste (mid-page) should be changed to indicate that high-level and TRU waste can be mixed (as well as low-level waste). **49-35**

All wastes should fit into high-level, TRU, low-level, mixed, hazardous, or non-hazardous waste categories. Management and disposal of "other wastes" would be problematic. **49-36**

Volume 2, Section C.2, page C-6
Why would the plutonium be further purified in a "Second Plutonium Cycle" ? Is it considered to be surplus plutonium? **49-37**

Volume 2, Section E.2.2, Table E-2, page E-9
The table should include units; all impacts are per person-rem. **49-38**

Volume 2, Section E.4.1, page E-15, 1st paragraph, line 5
"Department of Environmental Quality" should be "Division of Environmental Quality." **49-39**

Volume 2, Section F.2.2.1.2, page F-11, 2nd paragraph
This paragraph is not clearly written and is hard to follow. Also, what are the 0.03 units referred to in "all major systems are known to have survived the 0.03 Borah earthquake..."? **49-40**

Volume 2, Section F.2.2.1.3, page F-24, 1st paragraph
"The location of the F-Canyon facility is far away from any airport...." A distance should be listed to quantify "far away." **49-41**

Volume 2, Section G.5.5, page G-16
"(ANL 1994)" A more recent reference for vehicle accident and fatality rates is Saricks, C. L. and M. M. Tompkins, 1999, State-Level Accident Rates of Surface Freight Transportation: A Re-Examination, ESD/TM-150, Argonne National Laboratory, Argonne, Illinois. **49-42**

Volume 2, Section G.5.6.2, 1st line
"The release fractions for were taken from...." The missing word between "for" and "were" should be provided. **49-43**

Response to Commentor No. 49 (Cont'd):

49-41: DOE Standard 3014-96 discusses the distances from where a facility could be affected by takeoff and landing accidents. F-Canyon is located outside the farthest distance identified in the standard, more than 40 kilometers (25 miles) away from a major commercial airport. A clarification was added to the text.

49-42: The new transportation accident frequencies from this reference have been incorporated into the EIS.

49-43: The text in section G.5.6.2 of Appendix G has been revised for clarity.

Commentor No. 49: Kathleen E. Trever (Cont'd)

REFERENCES TO COMMENTS

Bartholomay, R.C., B.T. Tucker, D.J. Ackerman, and M.J. Liszewski, 1997, Hydrologic conditions and distribution of selected radiochemical and chemical constituents in water, Snake River Plain aquifer, Idaho National Engineering Laboratory, Idaho, 1992 through 1995: U.S. Geological Survey Water-Resources Investigations Report 97-4086 (DOE/ID-22137), 57 p.

Berenbrock, C., and L.C. Kjelstrom, 1998, Preliminary water-surface elevations and boundary of the 100-year peak flow in the Big Lost River at the INEEL, Idaho: U.S. Geological Survey Water-Resources Investigations Report 98-4065 (DOE/ID-22148), 13 p.

Hackett, W.R., and Smith, R.P, 1994, Volcanic hazards of the Idaho National Engineering Laboratory and adjacent areas: INEL-94/0276, 31 p.

Jackson, S.M., I.G. Wong, G.S. Carpenter, D.M. Anderson, and S.M. Martin, 1993, Contemporary seismicity in the eastern Snake River Plain, Idaho based on microearthquake monitoring: Bull. Seis. Soc. of Amer., v. 83, no. 3, pp. 680-695.

Should you have any questions regarding these comments, please contact Robert Guenzler at (208) 528-2600.

Sincerely,

Kathleen E. Trever

Kathleen E. Trever
Coordinator-Manager

KET/ds

Response to Commentor No. 49 :

Commentor No. 50: Bennett Ramberg

COMMITTEE TO BRIDGE THE GAP
1637 BUTLER AVENUE, SUITE 203
LOS ANGELES, CALIFORNIA 90025
(310) 478-0829

September 27, 1999

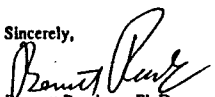
Susan Lesica
NE-40
U.S. Department of Energy
19901 Germantown Rd.
Germantown, MD 20874

Dear Ms. Lesica:

Enclosed please find comments on the Draft EIS for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel.

Thank you for your consideration.

Sincerely,



Bennett Ramberg, Ph.D.
Director of Research

Response to Commentor No. 50:

Commentor No. 50: Bennett Ramberg (Cont'd)

Comments Submitted by
Bennett Ramberg, Ph.D.
Director of Research,
Committee to Bridge the Gap
1637 Butler St.
Los Angeles, CA 90025
310 478-0829

On the "Draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel"

According to the dictionary, a "contortionist" is one who can twist his body into unnatural positions. Reading the "Nonproliferation Impacts Assessment for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel" (NIA)--an addendum to the "Draft Environmental Impact Statement for The Treatment and Management of Sodium-Bonded Spent Nuclear Fuel" (DEIS)--one comes away with the feeling that the Department of Energy (DOE) has adopted a contortionist's rationalization of reprocessing of spent nuclear fuel at the Argonne National Laboratory-West. (ANL-W). In so doing, the authors challenge a foundation of U.S. nuclear nonproliferation policy while laying the basis for implementation of more proliferation resistant actions.

In the commentary that follows, I intend to raise serious questions about the impact of EMT on the U.S. government's nuclear nonproliferation policy. In support of my concerns, I have enclosed the declarations of leading experts that appeared in litigation brought to halt the the Argonne National Laboratory-West's Electromeallurgical Treatment Research and Demonstration Project. 50-1

The following NIA "conclusion" provides a point of departure to illustrate the problem: "All but one alternative--the one involving PUREX reprocessing at SRS--are *fully consistent* with U.S. policy with respect to reprocessing and nonproliferation." (Page ES-7; emphasis added.) Note, however, that on page ES-6, DOE qualifies this affirmation declaring that the "consistent" alternatives "have the potential to raise limited concerns." In this Clintonesque word game, it is difficult to quantify how "limited" "limited concerns" are.

The report provides evidence that "concerns" are more than limited. On page ES-6 the authors note that EMT "could cause countries to question the U.S. commitment against reprocessing and *provide encouragement* for the expansion or initiation of reprocessing programs in other countries." (emphasis added) This is a curious consequence for a program that DOE purports to be "consistent "with Washington's nonproliferation policy. The body of the NIA supports this skepticism.

Response to Commentor No. 50:

50-1: The assessment of nonproliferation impacts is not part of the scope of the EIS. As stated in the Nonproliferation Impacts Assessment the alternatives involving PUREX reprocessing and broad application of electrometallurgical treatment of both driver and blanket fuel have a greater potential to provide encouragement to other countries to engage in plutonium reprocessing. Given the small quantity and unique characteristics of the sodium-bonded spent nuclear fuel and the reason for the treatment, however, such encouragement, if any, would be limited. Sodium-bonded spent nuclear fuel represents approximately 2 percent of the DOE's spent nuclear fuel inventory.

Commentor No. 50: Bennett Ramberg (Cont'd)

The contortions manifest early in the defense of EMT when the NIA cites ANL-W's contention, namely that IFR pyroprocessing "was designed to be 'proliferation resistant.'" (Page 3-4.) The NIA itself subsequently casts doubt citing no less and authority than the National Academy of Sciences: "Although the developers of the electrometallurgical technique argue that the technology is proliferation resistant, any [spent nuclear fuel] processing approach that is capable of separating fissionable materials from associated fission products and transuranic elements could be redirected to produce materials with nuclear detonation capability." (Page 3-5)

The contortions continue in another direction when the NIA concludes on page 3-8 that EMT in its "current equipment configuration" is incapable of producing weapons useable plutonium. At the same time DOE--echoing the National Academy of Sciences--concedes that "additional steps" could be taken to permit the procurement of weapons useable plutonium. In this regard the NIA notes that the compact EMI process allows for a "more concealed plutonium purification process." (Page 3-14.) It acknowledges that EMT will extract weapons-useable HEU. On page 3-14 the authors note diversion scenarios for both nuclear weapons useable materials.

50-1

Before proceeding into a comparative evaluation of spent fuel treatment alternatives, the NIA strains to distinguish EMT from reprocessing. Civil reprocessing has been contrary to public policy since the Ford Administration. The Clinton Administration has drawn an artificial line declaring that reprocessing refers solely to the extraction of plutonium. This is contrary to the common definition of reprocessing which is the separation of spent fuel into its constituent parts. (See the critical reviews of how reprocessing has been represented at ANL-W in Attachments 1 and 2 by Thomas Cochran, Ph.D. and Professor Albert Wohlstetter. Ph.D., respectively.)

Recognizing the dilemma, the authors engage in remarkable contortions attempting to suggest that EMT is not reprocessing. In this view, "Because EMT is not capable of separating plutonium from fission products, it cannot be considered plutonium reprocessing." In the same paragraph, however, the NIA concedes that "separation of weapons-usable plutonium is possible using a modified EMT electrorefiner in a hybrid system." With regard to HEU, the authors concede that extraction from spent fuel is "reprocessing": "EMT does recover HEU from HEU-containing spent nuclear fuel, similar to other DOE reprocessing facilities such as the Idaho Chemical Processing Plant and the

Response to Commentor No. 50:

Commentor No. 50: Bennett Ramberg (Cont'd)

H-Canyon facility at the Savannah River Site (SRS), and for that reason, EMT could be recognized as a reprocessing technology." (Page 3-15.)

As a result of this the NIA concludes, "Because of the similarities between EMT and conventional reprocessing, in particular the ability of EMT to recover HEU and the role of the EMT electrorefiner in a potential hybrid plutonium recovery process, both domestic and export applications of EMT pose concerns with respect to U.S. nonproliferation policy." (Page 3-15.) The authors punctuate this point on page E-16 noting, "*Except in cases where EMT exhibits a decisive advantage (e.g., in security, cost, environmental, or health and safety) over other alternatives, the use, export, development, or promotion of this technology could cause countries to question the U.S. commitment against reprocessing. Closely scrutinizing proposals for applying EMT (and similar fissile material separations technologies) will help mitigate this issue.*" (italics included) This is a peculiar conclusion for a technology that DOE contends is "consistent" with nonproliferation. Furthermore the authors fail to define what "scrutiny" is required to address their concerns.

All this is prolog to a curious comparative evaluation of the proliferation resistance of various sodium bonded spent fuel disposal alternatives. The Department of Energy argues that save for PUREX, all remaining alternatives are "acceptable in terms of nonproliferation risk." (Page ES-6.) However, Chapter 6's assessment demonstrates quite the contrary: some alternatives are clearly less risky than others and, it is not at all clear, as the report concludes, that the benefits of some are simply "marginal." (Page ES-6.)

50-1

Table 6-1 provides a point of departure. The authors indicate that in four categories EMT "could raise proliferation concerns." The text provides elaboration repeating concerns raised elsewhere. DOE concedes that it is difficult to assure application of international safeguards to EMT because bulk processing of nuclear material and separation of fissile material. Furthermore,

> "The similarities between EMT and conventional reprocessing would have somewhat greater potential to encourage reprocessing in other countries than would the high-integrity cans or melt and dilute options. This potential stems primarily from its ability to produce weapons-useable HEU and the historical origins of EMT as part of the IFR breeder fuel-cycle technology, which can be perceived as having several parallels to the PUREX technology used worldwide to process spent nuclear fuel. Extending the time that U.S. separations facilities operate and using a separations process to prepare spent nuclear fuel for geological disposal (while at

Response to Commentor No. 50:

the same time acknowledging that the fuel does not pose near-term safety and health vulnerabilities and that such processing technically is not required) could serve to undermine U.S. credibility in expressing concern to other countries about the proliferation problems associated with conventional reprocessing in the nuclear fuel cycle." (Page 6-4.)

The authors then proceed to rationalize EMT through a serious of contortions as the paragraph continues. They argue the technology is not intended to extract plutonium. Note however that the paragraph concedes that EMT has parallels to PUREX which is a plutonium reprocessing technology. The report then argues that EMT is necessary to address the "unique chemical reactivity requirements of the highly unusual type of spent fuel." However, the authors grant that other technologies which do not require reprocessing can cope with the challenge. A third defense of EMT argues that the technology intends to prepare the fuel for disposal rather than breeding. This is a *non sequitur*; with the conclusion of operations at EBR II, there is no breeder program in the United States. Finally, the authors acknowledge that that EMT would challenge verification under the Fissile Material Cutoof Treaty.

Were EMT the only alternative apart from PUREX, perhaps a case could be made for its application. However the NIA reports alternatives. High Integrity Cans for blanket assemblies and Melt and Dilute both reduce the proliferation challenge. The No Action alternative, defers the problem for an interim period. (In Attachment 3, Professor Frank von Hippel, Ph.D. argues that there is no urgency to treat the spent fuel at ANL-W.) Indeed, No Action allows time for development of "less mature" proliferation resistant technologies mentioned in the NIA and the DOE/EIS, e.g. GMODS and plasma arc methods. (Pages 6-9 of the NIA and S-23 and 2-31 of the DEIS. See also Professor James Warf, Ph.D., Attachment 4, on alternatives to extract sodium from the spent fuel without reprocessing.)

50-1

The NIA's demonstration that practical alternatives exist that do not bear the proliferation onus of EMT, coupled to others in development, begs the following questions: Why is EMT being promoted? Why has DOE engaged in the noted contortions? Why can't DOE await the maturation of promising technologies under development which do not raise the proliferation specter?

In conclusion, the foregoing analysis demonstrates that EMT is contrary to the "major principals" laid out in the September 27, 1993 White House "Nonproliferation and

Commentor No. 50: Bennett Ramberg (Cont'd)

Export Control Policy Statement" which declares "Our national security requires us to accord higher priority to nuclear nonproliferation." The application of EMT, which DOE acknowledges is reprocessing and could "provide encouragement" for reprocessing in other countries, is clearly inconsistent with the "Statement." Accordingly, DOE should halt the application of EMT without delay.

50-1

Attachments

Attachment 1: Declaration of Thomas B. Cochran, Ph.D.

Attachment 2: Declaration of Albert Wohlstetter, Ph.D.

Attachment 3: Declaration of Frank Von Hippel, Ph.D.

Attachment 4: Declaration of James C. Warf, Ph.D.

Response to Commentor No. 50:

Commentor No. 51: Charles Rice

Citizens Advisory Board

Idaho National Engineering and Environmental Laboratory

99-CAB-099

September 28, 1999

Susan Lesica
NEPA Document Manager
U.S. Department of Energy
Office of Nuclear Facilities Management
Office of Nuclear Energy Science, and Technology, NE-40
19901 Germantown Road, Germantown, Maryland 20874-1290,

Attention: DOE/EIS-0306

Dear Ms. Lesica:

Note: The Site-Specific Advisory Board (SSAB) for the Idaho National Engineering and Environmental Laboratory (INEEL), also known as the INEEL Citizens Advisory Board (CAB), is a local advisory committee chartered under the Department of Energy's (DOE) Environmental Management SSAB Federal Advisory Committee Act Charter.

The Idaho National Engineering and Environmental Laboratory Citizens Advisory Board (INEEL CAB) reviewed the *Draft Environmental Impact Statement (EIS) for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel* and two companion documents, the *Cost Study of Alternatives Presented in the Draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel* and the *Nonproliferation Impacts Assessment for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel.*

The enclosed recommendation, INEEL CAB Recommendation #64, conveys the INEEL CAB's comments on the three documents. The recommendation was reached through consensus at our September 1999 meeting. I might add that we appreciated your willingness to extend the public comment period to allow our participation.

We await your response to this recommendation.

Sincerely,

Charles M. Rice
Chair, INEEL CAB

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Response to Commentor No. 51:

Commentor No. 51: Charles Rice (Cont'd)

cc: Dieter Knecht, INEEL CAB Spent Nuclear Fuel Committee Chair
Beverly Cook, DOE-ID
Greg Bass, DOE-Chicago
Carolyn Huntoon, DOE-HQ
Bill Magwood, DOE-HQ
Martha Crosland, DOE-HQ
Fred Butterfield, DOE-HQ
Larry Craig, U.S. Senate
Mike Crapo, U.S. Senate
Mike Simpson, U.S. House of Representatives
Helen Chenowith, U.S. House of Representatives
Laird Noh, Chair, Idaho Senate Resources and Environment Committee
Golden C. Linford, Chair, Idaho House of Representatives Resources and Conservation Committee
Jack Barraclough, Chair, Idaho House of Representatives Environmental Affairs Committee
Gerald Bowman, DOE-ID
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Response to Commentor No. 51 (Cont'd):

Commentor No. 51: Charles Rice (Cont'd)

Citizens Advisory Board
Idaho National Engineering and Environmental Laboratory

Draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel

The Idaho National Engineering and Environmental Laboratory Citizens Advisory Board (INEEL CAB) has reviewed the *Draft Environmental Impact Statement (EIS) for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel* and two companion documents, the *Cost Study of Alternatives Presented in the Draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel* and the *Nonproliferation Impacts Assessment for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel*. This recommendation, reached through consensus, presents our comments on the three documents.

During the scoping period for this EIS, the INEEL CAB recommended that DOE consider the possibility of using different treatment processes for the driver fuel and the blanket fuel. We **commend DOE for taking that recommendation to heart.** We also feel that DOE was responsive to other recommendations we made during the scoping period, including requests for (1) a listing of all assumptions and (2) bounding estimates of shipments in and out of Idaho and estimates of storage duration(s). The following recommendations address our prior recommendations that were less well addressed and other concerns that arose during review of the draft EIS. 51-1

During the scoping period for this EIS, the INEEL CAB recommended that DOE evaluate the impacts of additional alternatives. We appreciate that DOE accepted that recommendation. **The INEEL CAB recommends that DOE give more consideration to the Glass Material Oxidation and Dissolution System and the Direct Plasma Arc-Vitreous Ceramic Treatment process in the Final EIS.** 51-2

The INEEL CAB recommends that DOE construct one more alternative and evaluate the impacts of that alternative in the Final EIS. The additional alternative should entail taking no action for the driver fuel. The components of this additional alternative are presented in other alternatives considered. Presentation of the impacts of these components separately (in different configurations) does not allow the public to evaluate this particular combination. This alternative would allow DOE more time to develop other non-separation technologies for possible treatment of the driver fuel and to allow further development work to determine the feasibility of removing sodium from the driver fuel (which would thereby allow disposal in High Integrity Cans). The INEEL CAB is not recommending selection of this additional alternative at this time, but would like to evaluate the impacts of such an alternative in comparison with those presented in the EIS. 51-3

During the scoping period for this EIS, the INEEL CAB recommended that relevant documents be made available during this comment period to support an informed public review of the Draft EIS. We were pleased to receive the *Cost Study of Alternatives Presented in the Draft* 51-4

Response to Commentor No. 51 (Cont'd):

51-1: DOE appreciates the commentor's commendation. DOE revised the scope of the EIS based on comments provided during the public scoping period.

51-2: The reasons why DOE dismissed the GMODS and direct plasma arc-vitreous ceramic treatment processes from its list of reasonable alternatives are provided in Section 2.6 of the EIS. There has been no new information since issuance of the draft EIS to change this position. Should DOE decide to take no action and wait for the development of a technology such as GMODS or the plasma arc process in its Record of Decision, additional NEPA documentation would be required to assess the impacts from the use of such technologies.

51-3: The environmental assessment of Alternatives 2, 3, 4, and 5 in Chapter 4 of the EIS presents the impacts from treatment of the driver and blanket sodium-bonded spent nuclear fuel separately. Conclusions on the environmental impacts of the alternative suggested by the commentor can be easily drawn, especially since the environmental impacts for all alternatives, including no action, are small and have been shown to not be a discriminator between alternatives. As discussed in Section 2.5 of the EIS, DOE will consider combinations of technologies, options, and fuel types, including combinations not included among the specific combinations considered in the EIS, in reaching its decision.

51-4: The National Academy of Sciences' National Research Council prepared nine reports on the electrometallurgical treatment technology that have been reviewed by DOE. These reports are located in the public reading rooms. The National Research Council completed its review of the electrometallurgical treatment technology in September 1999, and the final summary report on the Electrometallurgical Treatment Research and Demonstration Project was published in April 2000. DOE will consider the findings in this final report in determining the technical risk associated with the electrometallurgical technology alternatives in the EIS. Technical risk will be a factor in preparing the Record of Decision, which is scheduled for completion no sooner than 30 days after publication of the final EIS. Data generated during the demonstration project were used in preparing this EIS. Although NEPA does not provide for public hearings and a formal comment period following the issuance of a final EIS, the public is free to comment on the final document prior to publication of the Record of Decision.

51-5: Actual costs for treating and managing sodium-bonded spent nuclear fuel are not part of the EIS process. However, the Cost Study states that $47 million is the net present value of the disposal fee in 2000, i.e., the year

Commentor No. 51: Charles Rice (Cont'd)

Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel and the *Nonproliferation Impacts Assessment for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel*. Two other important and relevant studies still underway should have bearing on the decisions that this EIS will support. Inasmuch as the Draft EIS places primary emphasis on electrometallurgical treatment technology, the not-yet-reported electrometallurgical treatment demonstration project and the pending National Research Council's review of the electrometallurgical treatment process appear relevant. We regret that our review of the Draft EIS is less well informed than desired because the results of those two studies are not yet available. **The INEEL CAB recommends that DOE enhance public participation in this environmental review by allowing subsequent public comment period(s) once the other studies are available for public review.**

51-4 (Cont'd)

The *Cost Study of Alternatives Presented in the Draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel* presented relevant data on the alternatives considered in the Draft EIS. We noted, however, numerous apparent discrepancies and possible inaccuracies in the cost data presented. Such discrepancies and inaccuracies confuse the reader.

For example, Section 2.2 2 (pages 2-3 through 2-5) and Table 2-3 (on page 2-5) summarize costs associated with Alternative 1 by various cost elements. The text in Section 2.2.3 (on page 2-4) and the table both state that the net present value (in millions of Year-2000 dollars) for one cost element—waste form qualification at Argonne National Laboratory - West—would be $52 million. The cost estimate presented on Table 2-3 for another cost element—disposal fees for high-level radioactive waste—differs from the dollar value presented in the text, however. According to the table, disposal of high-level radioactive waste would cost $47 million; the text in Section 2.2.4 reports that the "repository fee" for 135 high-level radioactive waste disposal canisters would be "about $64 million" in 2015. There is no explanation for the difference between the two numbers. **The INEEL CAB recommends that DOE revise the cost study and that all cost estimates be presented in a readable and understandable form to support informed public review of environmental documentation.**

51-5

The INEEL CAB supports U.S. goals regarding nonproliferation. We recommend that DOE base decisions related to the management of sodium bonded spent nuclear fuel on a sound analysis of the potential nonproliferation impacts.

The members of the INEEL CAB differ significantly with regard to their opinions and perspectives on the current U.S. policy regarding reprocessing. As a result, we were unable to reach consensus on a recommendation regarding any particular alternative at this time. Those who support the current U.S. policy against reprocessing may not be able to support any alternative involving separations. Those who do not support the current policy may support alternatives involving separation technologies. Because we believe we represent the range of public opinions on this topic, the INEEL CAB appreciates DOE's current dilemma.

51-6

Response to Commentor No. 51 (Cont'd):

2000 value of the $64 million paid in the year 2015. In Section 1.4 of the Cost Study, the nominal escalation rate is defined to be 2.8 percent and the official discount rate provided by the Office of Management and Budget is 4.9 percent. The numbers are, therefore, consistent as stated. On page 1-7, the Cost Study explains the methodology used. Annual operating costs are provided in nominal, current year estimates except where life-cycle costs are noted.

51-6: The assessment of nonproliferation impacts is not part of the scope of the EIS. However, DOE's Office of Arms Control and Nonproliferation assessed the potential nonproliferation impacts that may result from each of the alternatives and technologies analyzed in this EIS. This analysis is presented in the Nonproliferation Impacts Assessment, which states that, for this specific application, all alternatives except PUREX processing at SRS are fully consistent with U.S. policy on reprocessing and nonproliferation. DOE welcomes public comments on nonproliferation issues and has received and responded to many comments on these issues during the public comment period on the draft EIS.

51-7: The SBSNF EIS was prepared in accordance with Council on Environmental Quality regulations (40 CFR 1500-1508) and DOE's NEPA-related regulations (10 CFR 1021) and procedures. As explained in the introduction to the EIS, the purpose of the EIS is to assess reasonable alternatives for the treatment and management of sodium-bonded spent nuclear fuel. As part of this assessment, as noted by the commentor, the EIS lists and describes the assumptions and methodologies used to evaluate environmental impacts. These assumptions and methodologies are consistent with the assumptions used in other related DOE EISs. The "related EISs" alluded to by the commentor, which are interdependent parts of a larger action as outlined in the Record of Decision for the Programmatic Spent Nuclear Fuel EIS (60 FR 28680), have been incorporated by reference and used, as appropriate, in the SBSNF EIS (see 40 CFR 1508.25(a)1(iii)). As a result of their publication, discussions on data and assumptions presented in the Yucca Mountain Draft EIS and the Idaho High-Level Waste and Facilities Disposition Draft EIS in particular have been expanded in the SBSNF EIS. The contributory effects of these other ongoing related NEPA actions at INEEL and SRS are evaluated as part of the cumulative impacts analysis for those sites (see Section 4.11 in the EIS). DOE acknowledges the commentor's opinion that the public

Commentor No. 51: Charles Rice (Cont'd)

The INEEL CAB recommends that DOE support vigorous debate regarding the environmental impacts of reprocessing, as well as the potential for terrorist or rogue military use of nuclear materials.

51-6

During the scoping period for this EIS, the INEEL CAB recommended that DOE include a list of all assumptions that provide the basis for the assessment of impacts associated with the various alternatives. While the Draft EIS provided a list of all assumptions, the INEEL CAB recommends that more information be provided on common data and planning assumptions used in related EISs and other environmental documentation. Our recommendation was not adequately addressed by simply providing the title and contents of other ongoing analyses. The public deserves an assessment of the data and assumptions to assure consistency and compatibility with other proposed actions.

51-7

During the scoping period for this EIS, the INEEL CAB recommended that the EIS describe how each alternative would address the waste acceptance criteria for resulting waste products destined for disposal at current and planned disposal facilities. In response to that recommendation, the Draft EIS states that existing preliminary criteria for spent fuel and high-level waste have been developed by DOE's Office of Civilian Waste Management and that the final criteria will be established by the Nuclear Regulatory Commission (NRC). The reference document cited in the draft EIS was the "Civilian Radioactive Waste Management System - Waste Acceptance System Requirements Document (WASRD), DOE/RW-0351, 1998." We note that the WASRD was revised in April 1999 to add criteria for high-level waste glass and for plutonium ceramic glass composite in addition to criteria for spent nuclear fuel and high-level waste. **The INEEL CAB recommends that the Final EIS be revised to incorporate the revised WASRD.**

51-8

The INEEL CAB further recommends that DOE begin to address the requirements that will be imposed by the waste acceptance criteria before the NRC licensing process begins. We understand that the criteria for the high-level waste glass and the plutonium ceramic glass composite (as incorporated in the current revised WASRD under Section 4.2.3.1 "Specific Acceptance Criteria for HLW") were developed in response to input regarding the likely characteristics of those waste forms. **The INEEL CAB recommends that DOE work to develop preliminary waste acceptance requirements for the wastes that will result from the treatment selected in the Record of Decision (ROD) for this EIS as soon as the ROD is issued.** In that manner, the characteristics of the likely wastes will be incorporated into future revisions of the WASRD before NRC develops the final waste acceptance criteria.

51-9

The INEEL CAB concluded that the Summary to the Draft EIS was overly brief and did not provide adequate explanations for the various alternatives evaluated nor for the impacts of those alternatives. We noted that the handout materials (provided at the public comment meetings on the Draft EIS) summarizing the alternatives and the impacts of those alternatives were reader-friendly and easily understood. **The INEEL CAB recommends greater reliance on reader-friendly formats in the Final EIS to help the public understand the information being presented.**

51-10

Response to Commentor No. 51 (Cont'd):

assessment beyond that already presented in the EIS is beyond the scope of this EIS.

51-8: As noted in both versions of the "Civilian Radioactive Waste Management System - Waste Acceptance System Requirements Document (WASRD), DOE/RW-0351, April 1999," the DOE spent nuclear fuel addressed by the Waste Acceptance System Requirements Document does not include the metallic sodium-bonded fuel addressed in the SBSNF EIS "which are candidates for treatment or processing prior to disposal." The EIS has been revised to identify the April 1999 version of the Waste Acceptance System Requirements Document. The analyses and results presented in the SBSNF EIS are not affected by the criteria identified by the commentor for high-level radioactive waste glass, plutonium ceramic glass composite, spent nuclear fuel, and other forms of high-level radioactive waste. DOE will determine the final waste acceptance criteria after the U.S. Nuclear Regulatory Commission issues its construction authorization, based on the successful demonstration of the safe, long-term performance of the repository in accordance with the U.S. Nuclear Regulatory Commission regulations.

51-9: The commentor's recommendation is noted. As stated in Section 2.7 of the EIS, DOE is actively working to develop final waste acceptance requirements for the waste discussed in this EIS. DOE expects the waste that would result from the alternatives analyzed in the EIS would be acceptable in a geologic repository.

51-10: DOE acknowledges the commentor's recognition of the usefulness of reader-friendly formats. The Summary to the EIS has been revised to incorporate a more reader-friendly format in illustrating the types of sodium-bonded spent nuclear fuel, the proposed action and alternatives, and the overall conclusions of potential environmental impacts presented in the handout materials.

Commentor No. 52: Edwin Lyman

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Comments on the Department of Energy's
Draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel

Nuclear Control Institute
September 28, 1999

The Nuclear Control Institute (NCI) appreciates the fact that the approach of the *Draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel* (SBDEIS) differs from that described in the Notice of Intent (NOI), and appears to reflect some of the comments made by NCI and others on the NOI.

In particular, the overall structure and title were changed so that electrometallurgical treatment (EMT) of the sodium-bonded spent nuclear fuel (SBSNF) could be evaluated on a more equal footing with the other treatment alternatives, including high-integrity canisters (HIC) and melt-and-dilute. 52-1

Also, the physical differences between the blanket and driver SBSNF assemblies were explicitly acknowledged, permitting the development of alternatives which utilize different approaches for the two types of fuel. (In this regard, we commend DOE for locating the documentation associated with the mechanical decladding and sodium removal of 7000 EBR-II blanket fuel in the 1980s that was reported to be missing at the NOI scoping hearing.) The wisdom of the approach has been confirmed by DOE's acknowledgment that "alternative technologies may have certain advantages (e.g. cost) for some or all fuel" (SBDEIS, Vol. I, pg. 2-41), and the fact that EMT of the entire SBSNF inventory has not been designated as the preferred alternative, as the NOI had envisioned. 52-2 52-1

Nonetheless, the SBDEIS, as well as the supporting cost and non-proliferation assessment documents, contain fundamental deficiencies and are utterly inadequate to the task of helping to determine a management approach for SBSNF which minimizes damage to the environment and to U.S. non-proliferation credentials. These deficiencies can be attributed in part to the relatively short time taken to produce these documents. We are greatly concerned by reports that the EIS process for SBSNF is being rushed so that funding can continue to flow to EMT development at Argonne National Laboratory (ANL)-West and the research team can be kept together. There 52-3

Response to Commentor No. 52:

52-1: The commentor's appreciation is noted. DOE revised the scope of the EIS based on comments provided during the public scoping period.

52-2: DOE appreciates the commentor's commendation.

52-3: The timing for this action is a programmatic issue rather than a safety issue. The SBSNF EIS was prepared in accordance with Council on Environmental Quality regulations (40 CFR 1500-1508) and DOE's NEPA-related regulations (10 CFR 1021) and procedures. Every effort was made to prepare an EIS that is complete and understandable. Further supporting documentation, such as the Cost Study and the Nonproliferation Impacts Assessment, is referenced and is available in DOE's public reading rooms. DOE is committed to improving its environmental management practices, to operating its facilities in a manner that meets or exceeds all applicable environmental, safety, and health requirements, and to cleaning up its environmental problems. The focus of the SBSNF EIS is to assess the potential environmental and health impacts associated with the treatment and management of sodium-bonded spent nuclear fuel. Although not final, the latest guidance provided by DOE's Office of Civilian Waste Management in their "Waste Acceptance System Requirements Document," Revision 3, April 1999 (see Section 4.12.1 of the EIS), indicates that it is highly probable that sodium-bonded spent nuclear fuel would not be acceptable in the repository without some stabilization and/or removal of the metallic sodium. The stabilization of the spent nuclear fuel and/or removal of the metallic sodium will provide greater protection of human health and the environment. Having completed the Electrometallurgical Treatment Research and Demonstration Project (see Section 1.6.3) and in planning the closure of its PUREX processing capabilities, DOE now needs to decide whether these processes are suitable for treating the remaining sodium-bonded spent nuclear fuel, or whether there is sufficient reason to delay a decision and wait for the development of other treatment technologies. Delaying the EIS could result in a loss of capability and of experienced, knowledgeable technical staff should DOE decide at a later date to use the electrometallurgical process to treat sodium-bonded spent nuclear fuel. Section 1.2 of the EIS has been revised for clarification. It is also worth noting that DOE has conducted four independent nonproliferation assessments of the electrometallurgical treatment technology over the last 11 years. These assessments have found the electrometallurgical treatment technology to be in accordance with U.S. nuclear nonproliferation policy

is no safety justification for making a hasty decision on such an important issue. 52-3 (Cont'd)

Comments on the *Draft EIS*

1. The need for EMT of the driver fuel has not been demonstrated.

The main problem with the SBDEIS is that it does not provide sufficient evidence that the SBSNF inventory will pose unacceptably high environmental risks if directly disposed of in a geologic repository. Without convincing evidence of this nature, it is not possible to conclude that the costs and risks of going forward with EMT are justified. At a minimum, to make a convincing case, DOE would have to show all three of the following:

1. Long-term chemical durability tests of SBSNF elements under repository-like conditions indicate that (a) the release rates of radionuclides are significantly greater than those of the proposed EMT waste forms or those of the much larger quantities of commercial SNF, vitrified high-level wastes and metallic uranium fuel (i.e. N-Reactor fuel) that will be placed in the repository; and/or (b) the presence of metallic sodium and uranium in the fuel results in chemical reactions of sufficient violence to cause significant structural damage to the repository, prolonged excessive heating or other undesirable changes. 52-4

2. The enhanced release of radionuclides, either directly from the SBSNF or indirectly as a result of damage to other waste forms in the repository, as well as other deleterious effects resulting from energetic chemical reactions, have significant impacts on the ability of the geologic repository to meet the regulatory performance criteria.

3. There are no technical remedies that could be applied to direct disposal of the SBSNF (i.e. local addition of a chemical buffer or special backfill) that could mitigate the risks associated with the presence of metallic sodium or uranium.

The SBDEIS contains no evidence along these lines, but merely asserts that the SBSNF might pose problems or "could complicate the process of certification of the SNF for disposal." This certification may well be complicated, but the cost and difficulty associated with certification must be compared to that which would be incurred by electrometallurgical treatment of this fuel.

Therefore, a variant of Alternative 2 should be considered in which the blanket fuel elements are mechanically declad and placed in high-integrity cans and the driver fuels are disposed of in high-integrity cans without any processing. The incremental impact of this option on repository performance compared to Alternative 2 should be evaluated. 52-5

2. Inconsistencies in the data must be corrected. 52-6

There are numerous inconsistencies in the technical data presented in the SBDEIS.

suitable for weapons production. DOE, in the Record of Decision, will take into account many factors besides this EIS and its supporting documents, including ongoing DOE programs, missions, and related, relevant NEPA actions. The commentor's opinion that the EIS and supporting documents may be deficient in supporting a decision is noted. DOE is confident that a sufficient amount of time was devoted to the preparation of this EIS and its associated documents.

52-4: As stated in the introduction to the EIS, the programmatic risk associated with implementing any of the potential alternatives for the treatment and management of sodium-bonded spent nuclear fuel, or with not treating this fuel, is the uncertainty surrounding the acceptability of DOE spent nuclear fuel for placement in a potential geologic repository. Although not final, the latest guidance provided by DOE's Office of Civilian Waste Management in its "Waste Acceptance System Requirements Document," Revision 3, April 1999 (see Section 4.12.1 of the EIS), indicates it is highly probable that sodium-bonded spent nuclear fuel would not be acceptable in a geologic repository without some stabilization and/or removal of the metallic sodium. The points raised by the commentor are the major reasons for uncertainties about the acceptability of this fuel. Performance of sodium-bonded spent nuclear fuel in a geologic repository depends on many factors (e.g., long-term fuel integrity, repository environment fuel/waste package survivability) and the presence of metallic sodium would complicate the modeling even further. Stabilization of the spent nuclear fuel and/or removal of the metallic sodium would provide greater protection for human health and the environment.

52-5: The alternative suggested by the commentor is similar, if not identical, to the direct disposal option of the No Action Alternative, which is evaluated in Section 4.2 of the EIS. It is not clear whether the commentor suggests the sodium is or is not removed before the blanket fuel elements are placed in high integrity cans. In either case, it is not the intent of this EIS to analyze the performance of a repository that would store spent nuclear fuel containing metallic sodium. This EIS assumes that the presence of metallic sodium in the sodium-bonded spent nuclear fuel may raise issues of acceptability in a repository and proposes technologies to either remove it or convert it into a nonreactive form to facilitate its disposal.

52-6: Since spent fuel degradation in storage cannot be ruled out, as described in Section 4.2.1 of the SBSNF EIS, air emissions under the No Action Alternative in the draft EIS were estimated using the adjusted values given

Commentor No. 52: Edwin Lyman (Cont'd)

Perhaps the most obvious one is the result (Table S-4) that the radiation dose to the public associated with the No Action alternative (SBSNF storage for 35 years) is nearly a factor of ten greater than that associated with Alternative 1 (EMT of the entire SBSNF inventory), although common sense dictates that the radioactive emissions resulting from processing SBSNF will exceed those resulting from storage. This curious result is a consequence of using an inconsistent set of assumptions for the two cases. At the public hearing in Arlington in August, DOE representatives assured the audience that this error would be remedied in the final EIS. 52-6 (Cont'd)

Another inconsistency is related to the categorization of the uranium recovered from the EMT process. The DEIS refuses to treat this material as a "waste," which is reflected in the tables listing the waste volumes resulting from the different alternatives. However, in the associated cost study, no credit is assigned to the recovered uranium. This is consistent with the expectation that this material will be "off-spec" and, moreover, that DOE has committed to tight restrictions on the sale of its surplus uranium for at least ten years in order to support the price of uranium originating from downblending of Russian HEU. However, a material without any value can properly be considered a waste and should be treated as such consistently through the documentation. 52-7

Comments on the *Nonproliferation Impacts Assessment for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel*

We are disappointed that DOE did not grant our request that the nonproliferation assessment for SBSNF treatment be provided to the public in draft form for comment as part of the EIS process. This is unfortunate, because the *Nonproliferation Impacts Assessment for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel* (NIA) provides only the most cursory and superficial examination of the important security issues associated with this action. As a result, the document in its current form sheds no new light on the situation and merely serves as a non-proliferation rubber-stamp. 52-8

1. The proliferation implications of other proposed applications of EMT must be considered.

Perhaps the most fundamental problem with the NIA is its refusal to look beyond the narrow constraints of the proposed program and consider the larger nonproliferation consequences of a decision by DOE to continue to pour resources into EMT development. While the NIA insists that "the Department has no current plan to use [EMT] ... beyond the potential treatment of the sodium-bonded spent nuclear fuel inventory," the authors are either dissembling or badly misinformed. One need go no further than the ANL-West World-Wide Web site page on EMT (http://www.era.anl.gov/spentfuel/emt.html) to discover that "the [EMT] process is being developed for application to all constituents of the DOE-owned spent nuclear fuel inventory" (emphasis ours). 52-9

Even worse, the NIA does not acknowledge the critical role played by EMT in DOE's

Response to Commentor No. 52 (Cont'd):

in the No Action Alternative for the Programmatic Spent Nuclear Fuel EIS. The adjustment was based on the ratio of the heavy mass inventory of the sodium-bonded spent nuclear fuel (60 metric tons) to the entire spent nuclear fuel inventory (274 metric tons) at INEEL. DOE assumed this estimate bounds any future degradation of the sodium-bonded spent nuclear fuel during storage at the INEEL site. The consequences resulting from this estimate were very small, and there was no intention to mislead the public. Since issuance of the SBSNF Draft EIS, DOE has modified the activities under both options of the No Action Alternative, as described in Section 4.2 of the final EIS; reevaluated the potential for sodium-bonded spent fuel degradation in wet and dry storage; and revised the estimates of air emissions and associated health effects. These new results are provided in the final EIS.

52-7: The uranium recovered from the electrometallurgical treatment process contains radioactive isotopes which render it unusable as surplus uranium without further processing to remove these impurities. DOE has not yet determined the final disposition of this uranium. For the purpose of the EIS, it is assumed that metal uranium ingots from the electrometallurgical treatment process would be stored in the Materials Building within the Zero Power Physics Reactor at ANL-W. The uranium recovered from the electrometallurgical treatment process has not been treated as a waste because of its potential value if it is further processed.

52-8: The SBSNF EIS has been prepared in accordance with NEPA, the Council on Environmental Quality regulations on implementing NEPA (40 FR 1500-1508), and DOE's NEPA implementation procedures (10 FR 1021). None of these require the preparation of a nonproliferation impacts assessment as part of the EIS process. As discussed in the introduction, the basic objective of this EIS is to provide the public and DOE decision-makers with a description of the reasonable alternatives for the treatment and management of DOE's sodium-bonded spent nuclear fuel and their potential environmental impact. DOE's Office of Arms Control and Nonproliferation separately assessed the potential nonproliferation impacts that may result from each of the alternatives and technologies analyzed in this EIS. The report stated that for this specific application all alternatives, except PUREX processing at SRS, are fully consistent with U.S. policy with respect to reprocessing and nonproliferation. DOE feels that this assessment provides the public with

Commentor No. 52: Edwin Lyman (Cont'd)

"Roadmap for Accelerator Transmutation of Waste (ATW). ATW is a profoundly misguided and dangerous effort, spearheaded by Senator Pete Domenici of New Mexico, to jump-start commercial spent fuel reprocessing in the United States with massive government subsidies. Although this program is clearly in violation of U.S. government policy on reprocessing, "it appears likely an ATW program will be established within the DOE in the near future, most probably under the auspices of the Office of Nuclear Energy," according to the ATW program manager, Greg Van Tuyle.[1]

According to Van Tuyle, "ATW separations are based on variations on the pyrometallurgical processing developed in support of the Integral Fast Reactor program," i.e. on EMT. Therefore, any meaningful assessment of the nonproliferation impacts of EMT development must contain an analysis of its function within the ATW plan, as well as the ramifications of ATW in an international policy context.

Specifically, the NIA should examine whether DOE's attempt to sanitize EMT from a non-proliferation perspective has led to the current confusion that has allowed ATW to go forward. Supporters of ATW only need to invoke DOE's numerous statements that EMT is a proliferation-resistant technology to defend their plan and argue that it does not violate the U.S. non-reprocessing policy. Meanwhile, there has been no analysis to show that a credible and cost-effective safeguards regime could be implemented on an ATW system based on EMT, or that EMT's purported proliferation-resistance would be meaningful in such a context.

52-9 (Cont'd)

It also should be noted that the current ATW strategy involves use of an aqueous process known as "UREX" to separate uranium from commercial oxide spent fuel prior to EMT processing.[2] Therefore, EMT process lines would be co-located with aqueous process lines. Since the NIA insists that EMT does not pose proliferation risks unless there is an aqueous separations capability nearby that could be used for further plutonium purification, the ATW strategy should raise serious concerns even according to the NIA's logic.

The refusal of DOE to produce a realistic and honest assessment of the proliferation concerns associated with EMT will have clear ramifications. A quixotic campaign to develop a massive ATW infrastructure in the U.S. will breath new life into faltering reprocessing industries all over the world. The alleged proliferation resistance of EMT will provide a fig leaf for countries like Japan, who are repeatedly faced with suspicions by their neighbors concerning their accumulation of plutonium in civil programs.

[1] Greg Van Tuyle, "The Roadmap for Accelerator Transmutation of Waste," *Nuclear Weapons and Materials Monitor* 3 (22), September 27, 1999, p.10.

[2] Greg van Tuyle, *op cit*

Response to Commentor No. 52 (Cont'd):

comments in response to the draft EIS, and the Nonproliferation Impacts Assessment will be among the factors considered during the decision-making process in preparing the Record of Decision.

52-9: This Nonproliferation Impacts Assessment analyzes the potential proliferation risks of all the alternatives presented in this EIS. Prepared by DOE's Office of Arms Control and Nonproliferation, the assessment concluded that for this specific application the electrometallurgical treatment process is fully consistent with U.S. policy with respect to reprocessing and nonproliferation. In the assessment, DOE acknowledges that future actions associated with the treatment and management of the sodium-bonded spent nuclear fuel should be closely scrutinized to evaluate their consistency with their individual and cumulative impact on U.S. policy concerning reprocessing and nonproliferation. While the commentor's concern about the proliferation implications of other proposed applications of electrometallurgical treatment is noted, these issues are beyond the scope of the SBSNF EIS.

52-10: There are several features of the electrometallurgical treatment process that make it adaptable to international safeguards. The process cell, made inaccessible to humans by high radiation, inert atmosphere, and thick concrete walls, has a minimal number of penetrations through which materials can be moved in and out. These openings are secured and can be readily monitored for material transfers. There are no liquid waste streams through which materials can be piped out of the facility. All by-products and waste from the process are in solid form, and thus are accountable by unit inventory. Finally, all materials moving out of the facility could be subjected to nondestructive examination if additional assurances were required under international safeguards agreements.

52-11: Although the assessment of nonproliferation impacts is not a part of the scope of the EIS, it should be noted that the residual highly enriched uranium in the cladding hulls can be determined accurately by several independent techniques. As much as 4 percent of the high enriched uranium in the EBR-II driver fuel may be left in the hulls to be disposed of as waste. Less than 1 percent of the depleted uranium would be left in the blanket fuel hulls because of different process conditions. Because the plutonium is preferentially dissolved from the blanket elements, no significant quantity of fissile material would remain in the blanket hulls. The blanket and driver hulls would be blended to reduce the enrichment of the residual uranium. Whether it would be desirable to blend a small amount of additional depleted

Commentor No. 52: Edwin Lyman (Cont'd)

2. A more detailed technical analysis of safeguard issues is essential.

The NIA glosses over many significant details. One crucial issue is the ability to apply accurate material accountancy techniques, both for domestic and international safeguards purposes. A highly inhomogeneous process such as EMT, with several different hard-to-characterize waste streams, will present significant challenges for material accountancy. While the NIA acknowledges that safeguards concepts for the complex bulk processing steps involved in EMT have not been demonstrated in detail, it asserts that effective international monitoring should be possible for a reasonable cost. However, it provides little justification for this assertion. 52-10

On the other hand, the NIA states that after EMT of SBSNF, "less than five percent of the uranium [remains] undissolved in the cladding hulls" (according to the National Academy of Sciences, material balances are about 95% for blanket assemblies and about 98% for driver assemblies). This is actually quite poor performance from a material accountancy perspective. For instance, over 3 MTHM of HEU is contained in the EBR-II driver fuel. Two percent left in the hulls equals over 60 kilograms of HEU (about two significant quantities). It is unclear how accurately the residual HEU content in the cladding can or will be measured. In addition, five percent of the 250 kilograms of "super-grade" plutonium in the blanket fuel left in the hulls exceeds 12 kilograms. The NIA does not explore the implications of these results for material accountancy. 52-11

The NIA also repeats without question the assertion that "pyroprocessing technology as envisioned in the IFR flowsheet is not capable of separating weapons-usable plutonium" because of the presence of uranium, radioactive fission products and minor actinides. This statement needs to be reevaluated in the current context for a number of reasons. First of all, as the current EMT demonstration project shows, over 95% of the uranium can be extracted on a steel cathode before extraction of the plutonium on a liquid cadmium cathode is attempted. Second, this author has shown that the residual fission products that remain with the plutonium after extraction provide a minimal radiation barrier, especially if the spent fuel has been out of the reactor for several months.[1] Third, the recently declassified statement by DOE that there are proliferation concerns associated with minor actinides, including Np-237 and Am-241, suggests that the cathode product may in fact have a greater utility for weapons than has previously been acknowledged, even without further purification. 52-12 52-13

The NIA also gives too much credit to the proliferation resistance of the EMT ceramic waste forms in concluding that there is a higher level of difficulty in recovering plutonium from them than from the original spent fuel elements. It is unclear why this is the case, since its analysis indicates that the radiation barrier provided by the waste forms will be no stronger than 52-14

[1] Lyman, E.S., "Interim Storage Matrices for Excess Plutonium: Approaching the 'Spent Fuel Standard' Without the Use of Reactors," PU/CEES Report No. 286, Center for Energy and Environmental Studies, Princeton, NJ, August 1994.

Response to Commentor No. 52 (Cont'd):

uranium in the metal waste in order to meet safeguards and waste disposal goals is still under evaluation as a part of Argonne's continuing waste form development program.

52-12: The commentor makes reference to the Integral Fast Reactor program. The purpose for the Integral Fast Reactor program was to develop an efficient, safe process for recycling nuclear fuel by using a liquid metal-cooled reactor in combination with an integral fuel reprocessing facility. As part of this program, the EBR-II was used for fuel-design and fuel irradiation testing. Congress cancelled funding for the Integral Fast Reactor program in 1994. The previously envisioned Integral Fast Reactor process is outside the scope of the EIS. The Nonproliferation Impacts Analysis states that the pyroprocessing technology as envisioned in the Integral Fast Reactor program is not capable of separating weapons-usable plutonium was based both on previous evaluations and the more recent results obtained from the electrometallurgical treatment demonstration project. The current demonstration has actually shown that greater than 99 percent of the uranium is dissolved from the blanket elements and an equal amount is deposited on the cathode prior to being scraped into a product collection container. However, in order for this process to work, the uranium concentration in the electrolyte must be maintained within a specified range. Uranium chloride is added in order to maintain the concentration of uranium in the electrolyte at a constant level through the fuel treatment campaign. There is no cadmium cathode nor is there a state of operations in which 95 percent of the uranium would be removed from the electrolyte. The unsuitability of the plutonium product from the modified Integral Fast Reactor program for weapons use is based on several physical characteristics in addition to its high radiation barrier.

52-13: The evaluation performed considered the entire mix of materials in the hypothetical cathode, including neptunium and americium. The quantities of neptunium 237 and americium 241 in the EBR-II blanket elements are quite small, and could not change the conclusions even if their consideration had been omitted from the evaluation.

52-14: Given sufficient time and resources, any chemical element can be separated from another. Alternative 3, PUREX processing at SRS, for example, is a fully developed process that has equipment and facilities that are capable

Commentor No. 52: Edwin Lyman (Cont'd):

that provided by the original spent fuel (in fact, according to the NIA the radiation barriers are exactly the same before and after processing the EBR-II fuel). No evidence is presented to support the assertion that recovery of plutonium from the ceramic waste form would be decisively more difficult than from the spent fuel elements.

52-14 (Cont'd)

In summary, the NIA's conclusion that EMT of SBSNF is consistent with U.S. policy on plutonium reprocessing is short-sighted and ignores the vastly expanded uses of EMT that are being considered by DOE. The worldwide ramifications of continued EMT development are far more serious and damaging to U.S. nonproliferation efforts worldwide than the NIA acknowledges.

Sincerely,

Edwin S. Lyman, PhD
Scientific Director

Response to Commentor No. 52 (Cont'd):

of separating plutonium from the blanket fuel elements. The recovered plutonium from this process, however, is addressed by the Surplus Plutonium Disposition EIS. For the complex chemistry of the electrometallurgical treatment ceramic waste form, processes, equipment and facilities would have to be developed to recover plutonium. Therefore, it is reasonable to conclude that plutonium recovery from this ceramic waste form would be more difficult than recovering plutonium from the sodium-bonded spent nuclear fuel and melt and dilute product.

Commentor No. 53: Richard Parkin

UNITED STATES ENVIRONMENTAL PROTECTION AGENCY
REGION 10
1200 Sixth Avenue
Seattle, Washington 98101

September 28, 1999

Reply To
Attn Of: ECO-088

Ref: 99-010-DOE

Sue Lesica
Office of Nuclear Facilities Management (NE-40)
Office of Nuclear Energy, Science, and Technology
U.S. Department of Energy
19901 Germantown Road
Germantown, MD 20874

Dear Ms. Lesica:

The Environmental Protection Agency has received and reviewed the draft Environmental Impact Statement (EIS) for the *Treatment and Management of Sodium-Bonded Spent Nuclear Fuel* consistent with our responsibilities under the National Environmental Policy Act and §309 of the Clean Air Act. The draft EIS examines six action alternatives to treat, contain, or treat and contain sodium-bonded spent nuclear fuel to facilitate its disposal in a geologic repository. The draft EIS does not identify a preferred alternative.

Based on our review, we have rated the draft EIS EC-2 (Environmental Concerns - Insufficient Information). This rating and a summary of our comments will be published in the *Federal Register*. We have enclosed a summary of the rating system used in our review for your reference.

It appears that the current drivers for this project are potential NRC regulations and the possibly inadequate current storage of spent nuclear fuels in high-integrity cans. However, the regulations remain unpromulgated and the viability of current storage of the spent fuels is not discussed in the EIS. Thus, the EIS fails to make a compelling argument that the proposed project is needed at this time. Moreover, the EIS was issued before final test results on the electrometallurgical process became available. This process is included in five of the six action alternatives. The absence of this information in the EIS prevents reviewers from fully assessing the efficacy of the the project.

Enclosed please find our detailed comments. We are interested in working closely with the Department of Energy in the resolution of these issues and I encourage you to contact Chris Gebhardt at (206) 553-0253 to discuss our comments and how they might best be addressed. Thank you for the opportunity to review this draft EIS.

Sincerely,

Richard B Parkin

Richard B. Parkin, Manager
Geographic Implementation Unit

Response to Commentor No. 53:

Commentor No. 53: Richard Parkin (Cont'd)

**Environmental Protection Agency Comments on the
Draft Environmental Impact Statement (EIS) for the
Treatment and Management of Sodium-Bonded Spent Nuclear Fuel**

Introduction

The Environmental Protection Agency (EPA) has completed its review of the draft EIS for the *Treatment and Management of Sodium-Bonded Spent Nuclear Fuel*. We are primarily concerned about the lack of information in the following areas. The EIS does not explain why immediate treatment is necessary. It would appear that treatment at this time might be premature since the NRC (Nuclear Regulatory Commission) has not yet promulgated regulations and a disposal site has not been chosen. Moreover, the need to replace the high-integrity cans has not been demonstrated. Finally, results of final testing of the electrometallurgical process that demonstrates the cost-and treatment-effectiveness of this method are not available in the EIS.

Purpose and Need

The EIS fails to make a compelling argument that the proposed project is needed. The EIS should describe in more concrete terms the conditions creating a need to treat sodium-bonded spent nuclear fuels. Disclosure of this nature would better meet NEPA's requirement of presenting accurate, high-quality information for public scrutiny (40 CFR 1500.1(b)).

Lack of Critical Information Establishing and Defining a Need

The EIS has identified a need to treat sodium-bonded spent nuclear fuels before sufficient information exists to demonstrate this. The EIS states that storage regulations could require treatment of these spent fuels and examines options to treat the fuels before the regulatory parameters defining the existence and the nature of this need have been established:

- the site decision as to whether to store the spent fuels over the long term, and if so, where (e.g., Yucca Mountain) has not yet been made;
- the Nuclear Regulatory Commission (NRC) has not yet promulgated regulations for the safe storage of spent fuels, nor authorized construction, for the still undecided site;
- it is unknown whether blanket fuels (which comprise 95% of the total spent fuels by weight) would be in compliance with the requirements of RCRA (Resource Conservation and Recovery Act) and NRC if stored in high-integrity cans after sodium removal; and
- DOE has not tested whether high-integrity cans can viably and safely store metallic uranium for the projected 100,000 years needed.

Presumably, site characteristics, general and site-specific regulations and the capabilities of waste storage equipment dictate acceptable waste standards and appropriate treatments to meet standards. This critical information should be developed before a decision on whether and how to treat spent nuclear fuels is made.

53-1

1

Response to Commentor No. 53 (Cont'd):

53-1: DOE's examination of options for the management and treatment of sodium-bonded spent nuclear fuel is based on the existing regulatory environment concerning long-term disposal of spent nuclear fuel and high-level radioactive waste. It is also based on the assumption that sodium-bonded spent nuclear fuel, as well as other DOE-owned spent nuclear fuel, would eventually be disposed of in a geologic repository, whether at Yucca Mountain or some other site. As stated in Section 1.2 of the EIS, DOE needs to reduce the uncertainties associated with qualifying sodium-bonded spent nuclear fuel for disposal so that the requirements of the State of Idaho Settlement Agreement and Consent Order are met, and disposal in a geologic repository is facilitated.

The Settlement Agreement calls for removal of all spent nuclear fuel from the State of Idaho by the year 2035. It would be environmentally prudent for the fuel at the time of removal to be in a form that is suitable for repository disposal, even if it is transported for continued storage to another site outside the State of Idaho.

The uncertainties associated with qualifying sodium-bonded spent nuclear fuel for repository disposal are based on the existing regulatory environment. As discussed in Section 4.12.1 of the EIS, one of the key U.S. Nuclear Regulatory Commission requirements for acceptance of spent nuclear fuel or high-level radioactive waste is that it cannot contain or generate materials that are explosive, pyrophoric, or chemically reactive (in a repository environment) in a form or amount that could compromise the repository's ability to perform its waste isolation function or to satisfy its performance objective (10 CFR 135(b)(1)). In addition, in accordance with the current version of the "Waste Acceptance Systems Requirements Document," issued in April 1999 by the DOE Office of Civilian Radioactive Waste Management, only spent nuclear fuel and high-level radioactive waste that is not subject to regulation under RCRA, Subtitle C, and meets all other acceptance criteria (e.g., packaging, uranium content), will be accepted for disposal. Although this determination for sodium-bonded spent nuclear fuel has not been made, it is a possible outcome. Based on the current regulatory environment, it is highly probable that sodium-bonded spent nuclear fuel will not be qualified for repository disposal without the removal or conversion of the metallic sodium to a nonreactive form.

The timing for this action is a programmatic issue rather than a safety issue. That is, the driver for the project is not "inadequate storage of spent nuclear fuel in high integrity cans," as the commentor appears to have concluded from the EIS. The EIS does not make this statement

Commentor No. 53: Richard Parkin (Cont'd)

No Need for Immediate Action

The information in the EIS does not demonstrate an urgent need to treat the spent nuclear fuels. The consent agreement between the State of Idaho and Department of Energy (DOE)/Navy specifies a time frame ending in the Year 2035 for removing the sodium-bonded spent nuclear fuels from Idaho. Presently, DOE is safely storing these spent fuels in high-integrity cans. Continuing to do so in Idaho for the short-term and other locations (e.g., Yucca Mountain) over the long-term does not appear problematic from the information presented in the EIS.

Insufficient Information about Electrometallurgical Treatment

The EIS calls for using electrometallurgical treatment in five of the six action alternatives. Nevertheless, the draft EIS was issued before final testing of the treatment- and cost-effectiveness of the electrometallurgical process were conducted. The results of these tests are absent from the draft EIS, and thus largely from public review and scrutiny. 53-1 (Cont'd)

Conclusion

We believe it critical that the draft EIS include the final tests results for the electrometallurgical process as well as information indicating the need for the project to achieve the dual purposes of NEPA: public disclosure and involvement and better decisions. Therefore, we suggest that you consider including the following information.

- the location of the geologic repository for the sodium-bonded, spent nuclear fuels,
- an explanation of the RCRA, AEA (Atomic Energy Act) and NRC standards that must be met for safe storage at the designated geologic repository,
- a description of the viability of high-integrity cans for the long-term storage of metallic uranium, and
- the test results of the cost- and treatment-effectiveness of the electrometallurgical process.

More Detailed Comments

Page S-3 53-2
Sidebars on this and other pages present background information in an appealing, useful way.

Page S-34: 53-3
In the last paragraph, the risk estimates should be "0.0088" vs "0.0088 x 10-4".

Section 4.4.4.1: 53-4
It should be noted somewhere in the evaluation that with regard to the EPA requirements for radioactive air emissions under 40 CFR 61, levels below the 10 mrem/year standard are acceptable and that lower levels are not necessarily "more acceptable".

Response to Commentor No. 53 (Cont'd):

Furthermore, the EIS does not assume that the sodium-bonded spent nuclear fuel is currently stored in high integrity cans. As stated in Section 1.2 of the EIS, DOE considers it prudent to evaluate the alternative technologies now, while DOE is performing site characterization activities for the potential repository at Yucca Mountain. Potential waste forms resulting from treatment or packaging of sodium-bonded spent nuclear fuel should be developed as much as possible in parallel with any repository development to promote consistency between the two efforts and to minimize the programmatic risks associated with waste qualification and acceptance for ultimate disposal. In addition, as discussed in Section 1.6.3 of the EIS, the Electrometallurgical Treatment Research and Demonstration Project was recently completed, successfully fulfilling all the criteria established at the outset of the project. In view of the results, DOE needs to decide whether electrometallurgical treatment is a viable technology for processing the rest of the sodium-bonded spent nuclear fuel, or whether some other process could offer environmental, cost, or nonproliferation advantages. Should DOE decide that electrometallurgical treatment is the appropriate treatment technology, the decision needs to be made while the facilities, skills, and personnel involved in the demonstration project are still available to carry out the treatment in an expedient and cost-effective manner. Section 1.2 of the EIS has been revised for clarification.

Final test results were made available in August 1999 and were used in preparing the EIS. The success criteria established at the outset of the project have been fulfilled. The environmental impact analysis associated with the electrometallurgical treatment process alternatives was based on actual data from the project. Section 1.6.3 of the EIS summarizes the current status and the results of the project.

DOE expects that spent nuclear fuel eventually will be disposed of in a geologic repository and this is a fundamental assumption made in the EIS. The site-specific characteristics of the potential repository are not expected to alter the uncertainties associated with qualifying sodium-bonded spent nuclear fuel for disposal. But even if one assumes that spent nuclear fuel will not be stored for the long term in a geologic repository, the treatment and management of this small quantity of spent nuclear fuel (2 percent of the total spent nuclear fuel inventory owned by DOE) to convert it to a stable, nonreactive form would be beneficial in any long-term storage environment. The high-integrity cans identified in the EIS protect the

Commentor No. 53: Richard Parkin (Cont'd)

Section 4.4.4.2:
If accidents are screened based on frequency (i.e., less than 1E-7), consideration should also be given to screening based on consequence (i.e., dose level). 53-5

Page 4-30:
The justification for excluding analysis of criticality accidents does not address the fact that although the frequency may be low, the dose consequences are likely greater than any other accident. This coupled with the understanding that a number of criticality accidents have occurred (including at INEEL), and the highly enriched nature of some of the uranium, would indicate the need for further evaluation. In addition, the issue of criticality is specifically mentioned in section S.1.1 with regard to geologic repository concerns so it seems inconsistent to exclude it from evaluation. 53-6

Page E-6, Radiation Protection Guides:
This section should include the EPA Radiation Protection Guidance to Federal Agencies for both occupational radiation exposure and exposure to the general public. 53-7

Page E-6, Limits of Radiation Exposure:
This section is misleading about the process and basis for radiation standards set by Federal agencies. ICRP and NCRP make recommendations. EPA takes those recommendations into account when issuing guidance (Radiation Protection Guidance for Federal Agencies). Federal agencies (including DOE) follow EPA Guidance in setting radiation standards under their own specific authority. 53-8

This section should include discussion of the EPA Radiation Protection Guidance to Federal Agencies for both occupational radiation exposure and exposure to the general public. These Guidance documents provide the basis for the implementation of radiation protection levels by other federal agencies (including DOE) under their own specific authorities.

This section should also include NCRP as a source of recommendations in addition to the ICRP.

Table E-1:

The proper "guidance criterion" associated with 4 mrem/year (drinking water) is the EPA drinking water standards referred to at 40 CFR 141. 53-9

This table should make it clear where exposure standards include an "ALARA" requirement and where they do not. For instance, the 5000 mrem/year and 2000 mrem/year worker exposure limits also require that exposures be maintained "as low as reasonably achievable". By contrast, the 40 CFR 190 and 40 CFR 61 standards are compliance criteria that do not require any additional effort to reduce exposures.

Response to Commentor No. 53 (Cont'd):

another barrier for protection in a repository environment; however, the barrier relied on to provide the isolation function at the repository is the waste package that would contain the standardized canisters which, in turn, would contain the high-integrity cans. In the environmental impact analysis, the Yucca Mountain Draft EIS takes no credit for the long-term integrity of either the standardized canisters or the cans (e.g., the high-integrity cans mentioned in this SBSNF EIS). Section 2.3.3 of the EIS has been revised to clarify the function of the high-integrity cans.

In the absence of metallic sodium, the other constituent of sodium-bonded spent nuclear fuel that is described as reactive and, in some cases, pyrophoric is metallic uranium. As discussed in Section 4.12.1, metallic uranium is defined under the Atomic Energy Act of 1954, as amended (42 U.S.C. 2001 et seq.), as a source, special nuclear, or by-product material and, therefore, is excluded from RCRA under 40 CFR 261.4(a)4. Furthermore, the purpose of the proposed action is to reduce the uncertainties associated with qualifying sodium-bonded spent nuclear fuel for repository disposal.

53-2: DOE acknowledges the commentor's recognition of the usefulness of the information presented in the sidebar format.

53-3: The commentor is correct; the risk estimate in the draft EIS should have been 0.0088.

53-4: DOE agrees with the commentor, and a clarifying statement has been added to Section E.2.1 of Appendix E, Limits on Radiation Exposure, in the EIS. This information had been addressed in Section 4.1.3.

53-5: To meet the Council of Environmental Quality's regulations, DOE's Office of NEPA Oversight has issued recommendations for the preparation of environmental assessments and environmental impact statements. In accordance with this guidance the analysis should identify a spectrum of the potential accident scenarios that could occur. The accident frequency should be "reasonably foreseeable." The primary purpose of accident analysis would be twofold: (1) to determine whether a proposed action has a potential for significant impact, and (2) to inform an agency (and the public) in making reasonable choices among alternatives. The accidents would have a likelihood of occurrence of greater than 10^{-7} per year. The guidance indicates that events with a probability of less than 10^{-7} will rarely need to be evaluated. Therefore, screening based on the frequency eliminates the need to evaluate the consequences.

Commentor No. 53: Richard Parkin

Response to Commentor No. 53 (Cont'd):

53-6: The potential for criticality could only exist if sufficient fissile material (enriched uranium fuel) existed in a favorable critical geometry. Operation of the hot cell facilities at ANL-W limits any moderator within the hot cell. The analysis of criticality accidents described in Section F.2.2.1.2 of Appendix F evaluated the potential for a criticality accident after a beyond design-basis earthquake, considering equipment operation at capacity and nuclear fuel staged for treatment, and concluded the likelihood of such an accident to be less than 10^{-7} per year. DOE evaluated an accidental criticality for melt and dilute processing of driver spent nuclear fuel. The consequences of such an accident are described in Appendix F and are summarized in Chapter 4 of the EIS. As indicated, the consequences to both the public and workers from a criticality accident in operations performed in the hot cells are very small. Once the fuel is put in a geologic repository, water could be available to potentially create a critical condition; therefore, criticality safety considerations would need to be implemented.

53-7: This section has been revised and clarifying statements have been added.

53-8: Clarifying statements have been added to Section E.2.1, of this EIS.

53-9: Clarifying statements have been added to Section E.2.1, of this EIS.

Commentor No. 54: Barbara Mathison

1335 Sew Creek Way
Meridian, Idaho,
83642
Sept. 9, 1999

Dear Ms. Susan Lesica,

I am commenting on the "Draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel"

I object to the Proposed Project at INEEL to Pyroprocessing spent nuclear waste. 54-1

This procedure separates out bomb-grade uranium from spent fuel and therefore runs counter to U.S. Nonproliferation goals. In addition, the technology can be modified to separate out bomb-grade plutonium 54-2

The project creates new forms of nuclear waste. It 54-3 54-4

Response to Commentor No. 54:

54-1: The commentor's objection to electrometallurgical treatment (pyroprocessing) of the sodium-bonded spent nuclear fuel at INEEL is noted.

54-2: The assessment of nonproliferation impacts is not a part of the scope of the EIS. However, none of the alternatives analyzed in this EIS would generate weapons-usable fissile materials at INEEL. Although highly enriched uranium is an interim product, it is downblended to low enriched uranium during electrometallurgical treatment (pyroprocessing). Within the current equipment configuration and design, it is not possible to produce weapons-usable plutonium by adjusting operating parameters. Traditional aqueous processing would have to be used after electrometallurgical treatment. However, traditional aqueous processing could also be used to produce weapons-usable plutonium directly from the spent nuclear fuel, without pyroprocessing.

54-3: All of the alternatives evaluated in this EIS would produce some forms of high-level radioactive waste. Electrometallurgical treatment (pyroprocessing) would produce two new waste forms, both of which are more stable than untreated sodium-bonded spent nuclear fuel. DOE expects these waste forms would be suitable for disposal in a geologic repository.

54-4: Congress determines how funds are allocated. DOE spends monies consistent with congressional direction. DOE is not in a position to make the difficult tradeoffs that may be required between alternative Federal programs and spending priorities. The issue of funding for the treatment and management of sodium-bonded spent nuclear fuel is beyond the scope of the SBSNF EIS.

54-5: In an effort to ensure that all interested parties had time to comment on the draft EIS, the deadline for transmittal of comments was extended from September 13, 1999, to September 28, 1999 (64 FR 49169).

54-6: Final test results were made available in August 1999 and were used in the EIS. The success criteria established at the outset of the project were fulfilled. The environmental impact analysis associated with the electrometallurgical process alternatives was based on actual data from the project. Section 1.6.3 of the EIS summarizes the status and the results of the project.

54-7: The National Academy of Sciences' National Research Council Committee prepared interim status reports on the results of the Electrometallurgical Treatment Research and Demonstration Project that have been reviewed

Commentor No. 54: Barbara Mathison (Cont'd)

takes valuable money away from greater environmental problems at INEEL.

It wastes taxpayer dollars.

INEEL has been featured twice on NBC's "Fleecing of America"

Please extend the comment period for 60 days. because DOE is missing the a) demonstration project results on pyroprocessing. b) National Academy of Sciences review of proposed treatment c) Cost analysis of various alternatives d) Nuclear weapons proliferation assessment by the D.O.E. e) the Yucca Mt. Environmental Impact Statement

Yours truly,

54-4 (Cont'd)

54-5

54-6

54-7

54-8

54-9

54-10

Response to Commentor No. 54 (Cont'd):

by DOE. The National Research Council completed their evaluation of the electrometallurgical treatment demonstration project in September 1999 and published their final summary report in April 2000. The final report findings will be considered during the decision-making process leading to the Record of Decision.

54-8: Environmental impact statements do not normally include a cost comparison between alternatives as costs are not environmental consequences. At the request of several members of the public during the Scoping Process for this draft EIS, DOE made a separate Cost Study available to the public during the comment period for the draft EIS. Copies of the Cost Study were mailed to individuals requesting the study, and copies were available during the four public hearings on the draft EIS.

54-9: Although the assessment of nonproliferation impacts is not a part of the EIS process, DOE's Nonproliferation Impacts Assessment was mailed out to interested members of the public on August 12, 1999 and is available by request. The assessment was also placed in DOE public reading rooms and distributed at public hearings during the public comment period on the draft EIS. Information from the assessment, along with other factors such as cost, schedule, environmental consequences, and technical risk will be considered during the decision-making process in preparing the Record of Decision.

54-10: The EIS has not specified a site for ultimate geologic disposal of waste, and thus is not affected by site-specific information that may be contained in the Yucca Mountain Draft EIS. As discussed in the revised Section 1.6.2.2 of this EIS, the Draft Yucca Mountain EIS was released by DOE in July 1999. Nothing contained in the Yucca Mountain Draft EIS changes the assumptions and the environmental impact analysis presented in the SBSNF EIS.

Commentor No. 55: Robert Bobo

Response to Commentor No. 55:

The SHOSHONE-BANNOCK TRIBES

FORT HALL INDIAN RESERVATION
PROJECT DIRECTOR (208) 478-3792
ENVIRONMENTALIST (208) 478-3709
SECRETARY (208) 478-3708
FAX (208) 237-0797

TRIBAL/DOE PROJECT
PIMA DRIVE
P. O. BOX 306
FORT HALL, IDAHO 83203

September 17, 1999

Ms. Sue Lesica
US Department of Energy, NE-40
19901 Germantown Rd.
Germantown, MD 20874

Dear Ms. Lesica:

RE: *DRAFT ENVIRONMENTAL IMPACT STATEMENT FOR THE TREATMENT AND MANAGEMENT OF SODIUM-BONDED SPENT NUCLEAR FUEL*

Attached are comments compiled by the technical departments of the Shoshone-Bannock Tribes. Thank you for the opportunity to review and comment on this document.

Sincerely,

Robert Bobo, Project Director

cc: Robert Pence, DOE ID American Indian Program Manager
file

Commentor No. 55: Robert Bobo (Cont'd)

COMMENTS TO THE DRAFT ENVIRONMENTAL IMPACT STATEMENT FOR THE TREATMENT AND MANAGEMENT OF SODIUM-BONDED FUEL

(all comments refer to the summary document)

Pg. iii - Instead of just saying there is a 45-day comment period, give the exact dates and deadlines. Also clearly indicate who should be the recipient of the comments. 55-1

Pg. S-2- Inasmuch as 98% of DOE's sodium bonded fuel is located at INEEL, it would seem prudent to perform any treatment at INEEL rather than ship the fuel to SRS or other facility. Transportation of SNF should be kept to a minimum. 55-2 55-3

Pg. S-2, sect S.1.1- One of the primary reasons given for treating sodium fuel is that "sodium reacts vigorously with water or moist air..." Why, then, is the sodium bonded fuel stored in water basins at INEEL and SRS? 55-4

Pg. S-13, sect S.3.1, 4th para. - What process is used to treat the uranium, and how will the uranium ingots be disposed? As waste or as a usable resource? 55-5

Pg. S-14, sect S.3.3, 1st para. - Please define "long-term". 55-6

Pg. S-14, sect S.3.3, last para. - Please expound further on "...designed to promote containment under repository conditions." What exactly would those containment criteria be? And what are the repository conditions? 55-7

Pg. S-14, sect S.3.4, 1st para. - It is this commentor's understanding that the only reason for treating sodium-bonded fuel is to remove the sodium; for disposal of SNF without sodium, no treatment will be necessary. If that is in fact the case, the first option of the Melt and Dilute Process seems unnecessary inasmuch as the sodium has already been removed (how?) before the remaining constituents are melted. If the sodium has already been removed, the mission has been accomplished, has it not? Dispose of the remaining constituents as non-sodium SNF that requires no treatment. Likewise, option two calls for treatment *after* the sodium has been removed. Again, if the sodium has already been removed, the goal set out in this EIS has been met. Why is it necessary to treat the remaining constituents? 55-8

Pg. S-15, sects S.3.5 and S.3.6, 1st para - Mention is made to research and development demonstration projects. What is the status of those projects? Have they demonstrated the feasibility of these two alternatives? 55-9

Pg. S-16, sect S.3.9, 1st para. - At the risk of belaboring the point, may I express perplexity in the statement, "For those methods that do not require the removal of metallic sodium prior to treatment..." Also, in the next paragraph, the statement "To remove the cladding after sodium has been extracted..." It seems totally superfluous to entertain any process for treatment *after* the sodium has been removed inasmuch as the removal of sodium is the sole reason given for treating the fuel to begin with. 55-8

Response to Commentor No. 55 (Cont'd):

55-1: The text cited by the commentor has been revised in the final EIS. In notices to the public published in the Federal Register, mailings to interested stakeholders, and in statements made by DOE at public meetings during the public scoping and comment periods members of the public were directed to submit comments to the DOE Document Manager, Ms. Susan Lesica.

55-2: The commentor's support for treating sodium-bonded spent nuclear fuel at INEEL, since most of it is located there, is noted. The environmental impacts from the transportation of blanket spent nuclear fuel from Idaho to SRS, discussed in Sections 4.9.4 and 4.9.6 of the EIS, are very small.

55-3: DOE assumes the commentor is referring to Alternatives 3 and 5, where the declad and cleaned (metallic sodium removed) blanket spent nuclear fuel would be transported to SRS for treatment. As explained in the EIS, the risks associated with fuel transport are very small. Regardless of the alternative, DOE would need to transport spent nuclear fuel and/or high-level waste out of INEEL. DOE will proceed in accordance with the DOE/Shoshone-Bannock Tribes Agreement-in-Principal, which covers notification and coordination of the transport of radioactive materials across the Fort Hall Reservation.

55-4: As discussed in Section E.4.6, the EBR-II fuel at INTEC's Basins 666 and 66 are stored inside sealed stainless steel cans that prevent the contact of basin water with the fuel cladding. During the average 17 years of storage in Basin 666, 10 of the 2,148 cans were confirmed to have water in-leakage. With water inside these cans, a fuel-water reaction produced hydrogen gas, which created bubbles that allowed detection of the water in-leakage. These observations are consistent with the fact that sodium and metallic uranium react with water to produce hydrogen and this is the reason that all the sodium-bonded spent nuclear fuel is stored in dry storage or sealed containers that prevent the exposure of the fuel cladding to water. The fuel at SRS is a single sodium-bonded spent nuclear fuel element encapsulated in an aluminum can, with no observed failure.

55-5: Two uranium stream products are produced by the electrometallurgical process. The uranium separated from the processed driver spent nuclear fuel would be diluted to about 19 percent uranium-235 (a low-enriched uranium fuel) before being cast into uranium ingots. Processing of the blanket spent nuclear fuel would produce depleted uranium ingots. As

Commentor No. 55: Robert Bobo

Response to Commentor No. 55: (Cont'd)

and actinide contamination (in trace quantities) that would require additional purification before they could be used commercially. Disposition of this surplus uranium will be the subject of a future NEPA review.

55-6: DOE interprets "long-term" to mean 1000 or more years after the repository's closure and no institutional control. The text in Section S.3.3 has been revised for clarification.

55-7: Containment criteria and repository conditions are provided in the Yucca Mountain Draft EIS and 10 CFR Part 60. Section S.3.3 of the Summary to this EIS has been revised for clarification.

55-8: As stated in the introduction to the EIS, the programmatic risk associated with implementing any of the potential alternatives for the treatment and management of sodium-bonded spent nuclear fuel, or with not treating this fuel, is the uncertainty surrounding the acceptability of DOE spent nuclear fuel for placement in a potential geologic repository. While DOE has drafted preliminary waste acceptance criteria for a geologic repository, the final acceptance criteria will be more refined. If the repository is developed, final acceptance criteria will not be available until after the U.S. Nuclear Regulatory Commission issues its construction authorization, based on the successful demonstration of the safe, long-term performance of the repository in accordance with the U.S. Nuclear Regulatory Commission regulations. As discussed in Section 1.2, the presence of metallic sodium is the primary, but not sole, reason for the proposed action. The presence of metallic uranium or the presence of highly enriched uranium could also complicate the process of qualifying the spent nuclear fuel for disposal. Such qualification would require sufficient data and predictive analyses to demonstrate that emplacement of the spent nuclear fuel would not adversely affect a repository's ability to protect the environment and worker and public health and safety. To ensure that the requirements of the State of Idaho Settlement Agreement and Consent Order are met, and to facilitate disposal, DOE needs to reduce the uncertainties associated with qualifying sodium-bonded spent nuclear fuel for disposal. Appropriate treatment and management of the sodium-bonded spent nuclear fuel (e.g., PUREX processing) would significantly reduce complications related to disposal qualification . The borosilicate glass waste form resulting from PUREX processing has been extensively tested and analyzed under conditions relevant to a geologic repository. It is expected that other waste forms (e.g., ceramic and metallic) would be suitable for repository disposal.

Commentor No. 55: Robert Bobo

Response to Commentor No. 55: (Cont'd)

55-9: The text in the draft EIS, as written, could imply that demonstration projects for the GMODS and plasma arc-vitreous ceramic processes are ongoing. This is not the case. The text has been revised to indicate that these technologies have the potential for treating both blanket and driver sodium-bonded spent nuclear fuel if it is demonstrated that they can deal with sodium and other factors.

Commentor No. 56: John Commander and Lowell Jobe

COALITION21 Supporting Tomorrow's Technologies With Facts + Not Fears!
P.O. Box 51232+Idaho Falls, Idaho 83405+208-528-2161+FAX: 528-2199

COMMENTS FROM COALITION 21 RE DOE/EIS-0306D DRAFT EIS for
SODIUM-BONDED SPENT NUCLEAR FUEL

It was nearly impossible for the public to evaluate the alternatives prior to having the independent cost and non-proliferation reports. With these finally available August 26th,we now submit the following comments: 56-1

1. Coalition 21 strongly supports the treatment of the sodium-bonded spent nuclear fuel (SBSNF) by the electrometallurgical process. The process should be used for both driver and blanket fuels, as described in Alternative 1 for the following reasons:

a. Since over 98% of the SBSNF is located at INEEL, it seems only reasonable that all of it be treated at ANL-W unless there was an overriding cost saving by using an alternative method.

b. ANL-W is the location of most of the experience and expertise in handling SBSNF materials.

c. The National Research Council in its most recent report expressed the opinion that, with the exception of the Purex process, all other alternatives to the electrometallurgical process were at an early stage of development. (Vol.1, p2-41). 56-2

d. Since the amount of SBSNF appears to be a fixed amount with no planned future additions, there is no further justification for funding the development of any other alternatives to handle the current amount of SBSNF.

e. Alternative 1, properly done, will demonstrate to the government and the public that the remnants of the Integral Fast Reactor program have been made ready for final disposal. It will have been accomplished in a timely manner with a technology compatible with the IFR concept. This position is consistent with the objectives of our lawsuit against DOE, which asks the court to require DOE to do a complete EIS on the disposal of the rest of the EBR-II reactor.

f. This alternative will also dispose of the sodium-bonded fuel, so that it cannot be used as an example of a failed technology by anti-nuclear groups.

2. We recommend that the cost report be redone and reissued to assure consistency in reporting, especially units of data tables. For example, in the separate cost report, Tables S-1 and 2 give cost summaries in 'millions of year 2000 dollars', while Table S-3 uses 'thousands of 2000 dollars (tabulated in tens of thousands) instead of using consistent 'millions of dollars'. To further confuse comparison of figures, Tables F-2 and F-3 thru F-9 list values as 'current dollars' (again tabulated in tens of thousands); this required searching for a clue to the discrepancy, found only in the bullet 2 under the F-2 Summary re Alternative 3's cost figure: 'at more than $130 million in 2009'. 56-3

Response to Commentor No. 56 (Cont'd):

56-1: The preparation of the Cost Study and Nonproliferation Impacts Assessment was expedited so that they could be mailed to interested parties on August 12, 1999, and be available to attendees at all of the public hearings on the draft EIS. Although these reports are not required for the EIS, they will be considered during the decision-making process in the preparation of the Record of Decision.

56-2: The commentor's support for using the electrometallurgical process to treat both driver and blanket fuel at ANL-W is noted. DOE acknowledges that the reasons provided by the commentor concerning the current location of the sodium-bonded spent nuclear fuel and the maturity of the electrometallurgical process are valid and have been the subject of discussion in the EIS. Issues such as funding or public relations are not within the scope of the EIS.

56-3: DOE believes the Cost Study provides the public with a reasonable comprehensive estimate of the cost of each alternative. There is no need to revise the Cost Study, because costs for treating and managing sodium-bonded spent nuclear fuel are not part of the EIS process. However, cost will be one of the factors considered in preparing the Record of Decision for the treatment and management of sodium-bonded spent nuclear fuel.

56-4: The costs presented in Table F-2 were discounted by the official discount rate provided by the Office of Management and Budget (4.9 percent) in accordance with the methodology described in Section 1.4 of the Cost Study. The ANL-W costs in Tables F-3 through F-9 are larger because they were not discounted, as stated on the last line of each table. The purpose of Tables F-3 through F-9 is to show the nominal costs in the year that those costs would be incurred.

56-5: The commentors' acknowledgment of the ranking of the estimated cost of alternatives as presented in the Cost Study is noted. Factors such as cost, schedule, environmental consequences, and technical risk will factor into the Record of Decision for the treatment and management of sodium-bonded spent nuclear fuel.

56-6: Tables S-3 and F-2 of the Cost Study are not numerically identical because the data in Table S-3 are discounted to year 2000 dollars, whereas the data in Table F-2 are in nominal dollars in the year in which the costs are incurred. From 2001 through 2006, Alternative 1 has lower annual costs than the other alternatives. The higher costs projected for Alternatives 4, 5 and 6 are partially explained by higher contingency factors that have been added to reflect their lesser degree of technological maturity.

Commentor No. 56: John Commander and Lowell Jobe

. Next, how can annual costs for 10 year summaries of data from Table F-2 be greater than the 35 year summaries of Table S-3 and the life cycle costs of alternatives 2,3 &4 of Table S-2?. Also, how can the annual ANL-W cost summaries from Tables F-3 thru F-9 be larger than ANL-W life cycle costs of Table S-2 except for the no-action and alternative 1? These inconsistencies need answering or correction. Any corrections might affect the other following comments. — 56-4

3. If cost were the only consideration, Alternative 2 for treating Driver-SNF and sodium removal plus packaging the blanket SNF in High-integrity cans at ANL-W would produces the lowest cost both by site and total cost, including waste disposal charges (Table S-2 of the cost report). Alternatives 1 and 3 were in second and third place, with Alternatives 4, 5 and 6 being substantially higher (Table S-2). — 56-5

4. The annualized cost tables S-3 and F-2, although not numerically identical, result in the same conclusions that would place Alternatives 4, 5, and 6 out of consideration, in line with their less mature status. Table F-2 also shows our recommended Alternative 1 for the elctrometallurgical treatment of all SBSNF as the lowest 10 year annualized cost. — 56-6

5. Although the Purex treatment part of Alternative 3 produces the least amount of high level waste (HLW), it produces several times as much Transuranic (TRU) and low level waste (LLW) as any other alternative. — 56-7

6. The No-Action alternative produces more HLW than any of the other alternatives. With an attendant cost of 73-86% of those for Alternatives 1 through 3 for no measureable solution to the problem, any consideration of this would be unwarranted. — 56-8

7. Since only the blanket-SBSNF can be handled at SRS and it must be first decladded and cleaned of sodium at ANL-W, the only advantage of Alternative 3 would be the transferal of that part of the SNF out of Idaho. There is no net time saving except for the Purex processing. — 56-9

8. All alternatives still require ANL-W to treat the driver SNF. Alternative 5 would still requires ANL-W to declad and clean the blanket-SBSNF of sodium prior to packaging it for shipment to SRS. In order to meet the 1995 Nuclear Waste Agreement with the state of Idaho, SRS would have to guarantee they could receive the material as it was readied, regardless of their prior commitments for handling other materials until 2035. — 56-10

9. We agree with the conclusions of the separate nonproliferation report that "Of the seven alternatives proposed in the Draft EIS, only one-that involving Purex reprocessing at the Savannah River Site raises significant nonproliferation issues....The alternative involving Purex reprocessing at SRS involves operation of a former weapons production facility and production of weapons-usable material." We see no non-proliferation problem with the electrometallurgical process. — 56-11

10. We recommend that the final EIS not be delayed to allow public comments on the final report of the electrometallurgical project by the Nat.Research Council of the NAS, a nationally recognized non-partisan organization; public comment is not required for this. Its report will be factored into the final EIS and ROD. — 56-12

Lowell A. Jobe

John C. Commander

Response to Commentor No. 56 (Cont'd):

56-7: The estimated waste generated by each of the alternatives is given in Table 2-4 of the EIS.

56-8: As indicated in the waste management sections of Chapter 4 of the EIS and summarized in Table 2-4, the direct disposal option of the No Action Alternative results in the highest volume of material (spent nuclear fuel or high-level radioactive waste) that would be disposed of in a repository. The commentor's opinion that the No Action Alternative should not be considered because it does not reduce waste volumes and the cost is nearly that of Alternatives 1 through 3 is noted.

56-9: Time-saving is one of the programmatic issues; however, the programmatic risk in implementing any of the potential alternatives for the treatment and management of sodium-bonded spent nuclear fuel, or of not treating this fuel, is the uncertainty surrounding the acceptability of DOE's spent nuclear fuel for emplacement in a potential geologic repository. While DOE has drafted preliminary waste acceptance criteria for a geologic repository, the final acceptance criteria will be more refined. If the repository is developed, final acceptance criteria will not be available until after the U.S. Nuclear Regulatory Commission issues its construction authorization based on successful demonstration of the safe, long-term performance of the repository in accordance with the U.S. Nuclear Regulatory Commission regulations. As discussed in Section 1.2, the presence of metallic sodium is the primary but not the only reason for the proposed action. The presence of metallic uranium, or the presence of highly enriched uranium could also complicate the process of certifying the repository. Such certification would require sufficient data and predictive analyses to demonstrate that placement of the spent nuclear fuel would not adversely affect a repository's ability to protect the environment and worker and public health and safety. To ensure that requirements of the State of Idaho Settlement Agreement and Consent Order are met and to facilitate disposal, DOE needs to reduce the uncertainties associated with qualifying sodium-bonded spent nuclear fuel for disposal. Appropriate treatment and management of the sodium-bonded spent nuclear fuel (e.g., PUREX processing) would significantly reduce the complications related to disposal qualification. The borosilicate glass waste form resulting from PUREX processing has been extensively tested and anlyzed under conditions relevant to a geologic repository. It is expected that other waste forms (e.g., ceramic and metallic) would be suitable for

Commentor No. 56: John Commander and Lowell Jobe

Response to Commentor No. 56 (Cont'd):

56-10: DOE agrees with the commentor that SRS should be able to receive declad and cleaned blanket fuel on or before 2035 for melt and dilute processing as soon as current missions are completed (around 2035). However, as indicated in Section 4.12.2, treatment at SRS could start as early as 2020 if additional treatment capacity becomes available, which is a programmatic rather than environmental issue.

56-11: The commentors' agreement with the Nonproliferation Impacts Assessment is noted.

56-12: The public comment period was extended from September 13 to September 28, 1999 (64 FR 49169) so that all interested parties would have additional time to comment on the draft EIS. While the results of the demonstration project were used to prepare the EIS, DOE agrees with the commentor that public comments on the final National Research Council report on the Electrometallurgical Treatment Research and Demonstration Project at ANL-W are not required by NEPA. It should be noted that the National Academy of Sciences' National Research Council Committee's interim status reports on the demonstration project were made available to the public in the public reading rooms.

Appendix B

Impact Assessment Methods

APPENDIX B
IMPACT ASSESSMENT METHODS

B.1 INTRODUCTION

This appendix briefly describes the methods used to assess the potential direct, indirect, and cumulative effects of the treatment and management of sodium-bonded spent nuclear fuel. Included are impact assessment methods for air quality; water resources; socioeconomics; waste management; and cumulative impacts. Each section is organized so that the affected resource is described first, and then the impact assessment method is presented. Methodologies were not developed for land resources; site infrastructure; noise; geology and soils; ecological resources; and cultural and paleontological resources, since impacts to these resources either would not occur or would be very small. This is because new construction would not be required, airborne and aqueous effluent would be controlled and permitted, and infrastructure requirements would not change for any of the treatment and management alternatives. Descriptions of the methods for the evaluation of human health effects from normal operations; facility accidents; transportation; and environmental justice are presented in Appendices E, F, G, and H, respectively.

Impact analysis varied with the resource area. For air quality, for example, estimated pollutant concentrations from the proposed facilities were compared with the appropriate regulatory standards or guidelines. Comparison with regulatory standards is a commonly used method for benchmarking environmental impacts and was done here to provide perspective on the magnitude of the identified impacts. The analysis of waste management impacts compared waste generated by the management of sodium-bonded spent nuclear fuel to the capacities of waste management facilities. Impacts in all resource areas were analyzed consistently; that is, the impact values were estimated using a consistent set of input variables. Also, similar presentations were developed to facilitate the comparison of alternatives.

B.2 AIR QUALITY

B.2.1 Description of Affected Resources

Air pollution refers to any substance in the air that could harm human or animal populations, vegetation, or structures, or that unreasonably interferes with the comfortable enjoyment of life and property. For purposes of this environmental impact statement (EIS), only outdoor air pollutants were addressed. These may be in the form of solid particles, liquid droplets, gases, or a combination of these forms. Generally, they can be categorized as primary pollutants (those emitted directly from identifiable sources) and secondary pollutants (those produced in the air by interaction between two or more primary pollutants, or by reaction with normal atmospheric constituents that may be influenced by sunlight). Air pollutants are transported, dispersed, or concentrated by meteorological and topographical conditions. Thus, air pollutant emission characteristics, meteorology, and topography affect air quality.

Ambient air quality in a given location can be described by comparing the concentrations of various pollutants in the atmosphere with the appropriate standards. Ambient air quality standards have been established by Federal and state agencies to allow an adequate margin of safety for protection of public health and welfare from the adverse effects of pollutants in the ambient air. Pollutant concentrations higher than the corresponding standards are considered unhealthy; those below such standards are considered acceptable.

The pollutants of concern are primarily those for which Federal and state ambient air quality standards ha been established, including criteria air pollutants, hazardous air pollutants, and other toxic air compounc Criteria air pollutants are those listed in 40 CFR 50. Hazardous air pollutants and other toxic compounds a those listed in Title I of the 1990 Clean Air Act, as amended; those regulated by the National Emissio Standards for Hazardous Air Pollutants; and those that have been proposed or adopted for regulation by t respective state or are listed in state guidelines. Also of concern are air pollutant emissions that may contribu to the depletion of stratospheric ozone or global warming.

Areas with air quality better than the National Ambient Air Quality Standards (NAAQS) for criteria pollutants are designated as being in attainment, while areas with air quality worse than the NAAQS for su pollutants are designated as being in nonattainment. Areas may be designated as unclassified when sufficie data for assigning attainment status are lacking. Attainment status designations are assigned by coun metropolitan statistical area, consolidated metropolitan statistical area, or portions thereof. Air Quality Cont Regions designated by the U.S. Environmental Protection Agency (EPA) are listed in 40 CFR 81.

For locations that are in an attainment area for criteria air pollutants, Prevention of Significant Deteriorati regulations limit pollutant emissions from new sources and establish allowable increments of pollut concentrations. Three Prevention of Significant Deterioration classifications are specified with the crite established in the Clean Air Act amendments. Class I areas include national wilderness areas; memorial par larger than 2,020 hectares (5,000 acres); national parks larger than 2,430 hectares (6,000 acres); and areas th have been redesignated as Class I. Class II areas are all areas not designated as Class I. No Class III are have been designated. Idaho National Engineering and Environmental Laboratory (INEEL) and the Savanr River Site (SRS) are within attainment areas (Class II) for the criteria air pollutants. INEEL is located abc 50 kilometers (33 miles) from the Craters of the Moon Wilderness Area Class I area. There are no Clas areas within 100 kilometers (62 miles) of SRS.

Baseline air quality is typically described in terms of the pollutant concentrations modeled for existing sourc at each site and the background air pollutant concentrations measured near the sites. For criteria pollutai at Argonne National Laboratory-West (ANL-W), baseline concentrations are based on 1) dispersion modeli at the site boundary centered at the INTEC facility, performed for the *Idaho High-Level Waste and Facilit Disposition Draft Environmental Impact Statement* (DOE 1999), using 1997 actual emissions and excludi ANL-W; and 2) dispersion modeling at the site boundary centered on ANL-W, using 1997 actual emissio The modeling performed for the High-Level Waste and Facilities Disposition EIS used EPA's ISCST3 mo with hourly meteorological data. The ANL-W modeling used EPA's SCREEN3 model, which is v conservative compared to ISCST3, and uses a set of worst-case meteorological conditions to predic maximum one-hour concentration. This one-hour concentration was converted to other averaging times us regulatory scaling factors (SCDHEC 1993). For these reasons, the ANL-W concentrations are extrem conservative. For SRS, concentrations for existing sources were obtained from the *Savannah River Site Sp Nuclear Fuel Management Final Environmental Impact Statement* (DOE 2000). These concentrations w compared with Federal and state regulations or limits (**Table B–1**). To determine human health risk, mode chemical concentrations in air were weighed against chemical-specific toxicity values.

B.2.2 Description of Impact Assessment

Potential air quality impacts of pollutant emissions were evaluated for each alternative. This assessm included a comparison of emissions from each alternative with applicable Federal and state ambient air qua standards. If both Federal and state standards exist for a given pollutant and averaging period, compliance evaluated using the more stringent standard.

Table B–1 Impact Assessment Protocol for Air Quality

Resources	*Required Data*		*Measure of Impact*
	Affected Environment	*Alternative*	
Criteria air pollutants and other regulated pollutants [a]	Modeled ambient concentrations (micrograms per cubic meter) of air pollutants from existing sources at site	Emission rate (kilograms per year) of air pollutants from facility and concentrations of air pollutants	Contribution of proposed alternative and total concentration of each pollutant at or beyond site boundary compared to applicable standard
Toxic/hazardous air pollutants [b]		Emission rate (kilograms per year) of toxic air pollutants from facility (micrograms per cubic meter)	

[a] Carbon monoxide; hydrogen fluoride; lead; nitrogen oxides; ozone; particulate matter with an aerodynamic diameter less than or equal to 10 microns; particulate matter with an aerodynamic diameter less than or equal to 2.5 microns; sulfur dioxide; total suspended particulates.

[b] Clean Air Act Title III pollutants, pollutants regulated under the National Emission Standards for Hazardous Air Pollutants, and other state-regulated pollutants.

Air pollutant emissions and concentrations data for each alternative, including the No Action Alternative, were based on information obtained in response to data requests to INEEL (ANL 1999) and on the SRS Spent Nuclear Fuel Final EIS (DOE 2000). For INEEL, a dispersion modeling analysis using the EPA SCREEN3 Model (Version 96043) (EPA 1995) was performed to estimate air quality impacts associated with the various alternatives. Emissions from ANL-W emergency diesel generators were modeled, in addition to cadmium emissions from the Fuel Conditioning Facility stack. The generators were modeled as ground-level volume sources; the cadmium emissions were modeled as an elevated point source release. Note that the emissions from the emergency generators are not specific to any given alternative, but are representative of the current operation of ANL-W. The cadmium emissions are specific to the current electrometallurgical treatment process. However, neither cadmium emissions nor emergency generator emissions are expected to increase as a result of any of the alternatives. Concentrations were predicted at 16 INEEL site boundary receptors and were compared to the ambient air quality standards.

For SRS, concentrations were obtained by scaling the concentrations in the SRS Spent Nuclear Fuel Final EIS (DOE 2000) based on the mass of sodium-bonded spent nuclear fuel to be processed under this EIS compared to the mass of spent nuclear fuel to be processed under the SRS Spent Nuclear Fuel Final EIS. The resulting concentrations were compared to the ambient air quality standards.

Ozone is typically formed as a secondary pollutant in the ambient air (troposphere). It is formed from primary pollutants such as nitrogen oxides and volatile organic compounds which emanate from vehicular (mobile), natural, and other stationary sources and mix in the presence of sunlight. Ozone is not emitted directly as a pollutant from the sites. Although ozone may be regarded as a regional issue, specific ozone precursors, notably nitrogen dioxide and volatile organic compounds, were analyzed as applicable to the alternatives under consideration.

Emissions of potential stratospheric ozone-depleting compounds such as chlorofluorocarbons were not evaluated, as no emissions of these pollutants were identified.

B.3 WATER RESOURCES

B.3.1 Description of Affected Resources

Water resources are the surface and subsurface waters that are suitable for human consumption; agricultu purposes; irrigation; or industrial/commercial purposes, and that could be impacted by the treatment of sodiu bonded spent nuclear fuel. This analysis involves the review of engineering estimates of expected water u and effluent discharges associated with the alternatives addressed in this EIS and the impacts of the alternatives on local water quality (including surface water and groundwater).

Surface water flow and quality data were obtained from existing reports. Groundwater users, information water use rights, and groundwater quality data also were obtained from existing reports.

B.3.2 Description of Impact Assessment

B.3.2.1 Water Use

The assessment of alternatives analyzed how the volumes of current water usage and effluent discharges wou change as a result of each alternative addressed in this EIS. A determination of the impacts of the alternati on water usage and effluent discharge is summarized in **Table B–2**.

Table B–2 Impact Assessment Protocol for Water Use and Effluent Discharge

Resources	*Required Data*		*Measure of Impact*
	Affected Environment	*Alternative*	
Surface water availability	Surface waters near the facilities, including average flow and numbers of downstream users	Volumes of withdrawals from and discharges to surface waters	Changes in availability to downstream users of water for human consumption, irrigation, or animal feeding
Groundwater availability	Groundwater near the facilities, including existing water rights for major water users and contractual agreements for water supply use within impacted area	Volumes of withdrawals from groundwater	Changes in availability of groundwater for human consumption, irrigation, or animal feeding

• For surface water availability, an impact is assumed if withdrawals exceed 10 percent of the 7-day, 10-year low-flow of the stre

If the determination reflected an increase in water use or effluent discharge, then an evaluation of the desi capacity of the water and effluent treatment facilities was made to determine whether the design capacity wo be exceeded by the additional flow. If the combined flow (i.e., the existing flow plus that of the propo activities) were less than the design capacity of the water and effluent treatment plants, then it was assun there would be no impact on water availability for local users, nor on the receiving stream from efflu discharges. Since flows from the facilities proposed to treat sodium-bonded spent nuclear fuel were found to exceed the design capacity of the existing water or effluent treatment facilities, no additional analysis water availability was performed.

B.3.2.2 Water Quality

The water quality impact assessment for this EIS analyzed how effluent discharges to surface water and groundwater resulting from the alternatives would affect current water quality. The determination of the impacts of the alternatives is summarized in **Table B–3**, and consisted of a comparison of the projected water quality with relevant regulatory standards such as the Clean Water Act, Safe Drinking Water Act, state regulations, and existing permit conditions. Separate analyses were conducted for surface water and groundwater impacts, as described below.

Surface Water Quality

The evaluation of surface water quality impacts focused on the quality and quantity of the effluent to be discharged and the quality of the receiving stream upstream and downstream from the discharge. The evaluation of effluent quality involved a review of the expected parameters, such as design average flows, as well as the effluent parameters reflected in the existing or expected National Pollutant Discharge Elimination System (NPDES) permit. Those parameters include metals; organic and inorganic chemicals; radionuclides; and any other parameters that affect the local environment. Water quality management practices were reviewed to ensure that NPDES permit limitations would be met. Factors that currently degrade water quality also were identified.

Table B–3 Impact Assessment Protocol for Water Quality

	Required Data		
Resources	***Affected Environment***	***Alternative***	***Measure of Impact***
Surface water quality	Surface waters near the facilities in terms of stream classifications and changes in water quality	Expected contaminants and contaminant concentrations in discharges to surface water	Compliance of discharges to surface water with relevant standards of Clean Water Act or with state regulations and existing NPDES permits
Groundwater quality	Groundwater near the facilities in terms of classification, presence of designated sole–source aquifers, and changes in quality of groundwater	Expected contaminants and contaminant concentrations in discharges that could reach groundwater	Concentrations of contaminants in groundwater exceeding standards established in accordance with Safe Drinking Water Act or state regulations

Groundwater Quality

No effluent discharges to groundwater are anticipated from any of the alternatives. Therefore, an analysis of impacts to groundwater quality was not performed.

B.4 SOCIOECONOMICS

B.4.1 Description of Affected Resources

Socioeconomic impacts are defined in terms of changes to the demographic and economic characteristics of a region. The number of jobs created by treatment of sodium-bonded spent nuclear fuel could affect regional employment, income, and expenditures. Job creation is characterized by two types: (1) construction jobs related to modification of existing facilities, which may be transient in nature and short in duration and thus less likely to impact public services; and (2) jobs related to plant operations that are required for a decade or more and possibly could create additional service requirements in the region of influence.

The socioeconomic environment is made up of two geographic regions, the regional economic area and the region of influence. Regional economic areas are made up of regional economies and include industrial and

service sector characteristics and their linkages to the communities within a region. These linkages determir the nature and magnitude of any effect associated with a change in regional economic activity. For exampl as work expands within a region, the money spent on accomplishing this work flows into the local econom where it is spent on additional jobs, goods, and services within the regional economic area.

Similarly, potential demographic impacts were assessed for the region of influence. The region of influen could represent a smaller geographic area—one in which only the housing market and local communi services would be significantly affected by a given alternative. Site-specific regions of influence we identified as those counties in which approximately 90 percent of the site's work force reside. This distributi reflects an existing residential preference for people currently employed at the sites, and was used to estima the distribution of new workers supporting the alternatives.

B.4.2 Description of Impact Assessment

The socioeconomic impact assessment analyzes both the potential positive and negative impacts of ea alternative, including the No Action Alternative. For each regional economic area, data were compiled on t current socioeconomic conditions, including unemployment rates, economic industrial and service sect activities, and the civilian labor force. Work force and cost requirements for each alternative were determin to measure their possible effect on these socioeconomic conditions. For each region of influence, cens statistics were compiled on population, housing demand, and community services. U.S. Census Bure population forecasts for the regions of influence were combined with overall projected work force requiremer for each of the alternatives being considered at each of the sites to determine the extent of impacts to housi demand and levels of community services (**Table B–4**).

B.5 Waste Management

B.5.1 Description of Affected Resources

The operation of support facilities for treating sodium-bonded spent nuclear fuel would generate several typ of waste, depending on the alternative. Such waste includes the following:

- **High-level radioactive:** The highly radioactive waste material that results from the processing of spe nuclear fuel, including liquid waste produced directly in processing and any solid waste derived from t liquid. High-level radioactive waste contains transuranic waste and fission products requiring permane isolation.

- **Transuranic:** Waste containing more than 100 nanocuries of alpha-emitting transuranic isotopes per gr of waste with half-lives greater than 20 years, except for: (1) high-level radioactive waste; (2) waste th has been determined by DOE and the EPA not to need the degree of isolation required by 40 CFR 191; a (3) waste that the U.S. Nuclear Regulatory Commission has approved for disposal, case by case, accordance with 10 CFR 61. Mixed transuranic waste contains hazardous components regulated under t Resource Conservation and Recovery Act (RCRA).

- **Low-level radioactive:** Waste that contains radioactivity and is not classified as high-level radioacti waste; transuranic waste; spent nuclear fuel; or the tailings or waste produced by the extraction concentration of uranium or thorium from any ore processed primarily for its source material. T specimens of fissionable material irradiated for research and development only, and not for the producti of power or plutonium, may be classified as low-level radioactive waste, provided the transura concentration is less than 100 nanocuries per gram of waste.

Table B–4 Impact Assessment Protocol for Socioeconomics

Resources	*Required Data*		*Measure of Impact*
	Affected Environment	*Alternative*	
Regional Economic Characteristics			
Work force requirements	Site work force projections from DOE sites	Estimated construction and operating staff requirements and schedule	Work force requirements added to site work force projections
Regional economic area civilian labor force	Labor force projections based on state population projections		Change in work force requirements as a percentage of the civilian labor force
Unemployment rate	1996 unemployment rates in counties surrounding sites and in host states		Projected change in unemployment rates
Population and Housing			
Population	Latest available population projection estimates from the U.S. Census Bureau	Estimated contribution to projected population	Projected change in population projection
Housing (percentage of occupied housing units)	Latest available rates from the U.S. Census Bureau	Assessment of potential need for housing units to meet work force requirements	Impacts are not expected since work force requirements would be small
Community Services			
Education Percentage of operating capacity for school districts in region of influence	Latest available rates from the U.S. Census Bureau	Assessment of potential need for new schools	Impacts are not expected since work force requirements would be small
Teacher-to-student ratio		Assessment of potential need for additional teachers	
Public safety Ratio of police and firefighters to 100,000 residents		Assessment of potential need for new officers and firefighters	
Health care Number of hospital beds and physicians per 100,000 residents		Assessment of potential need for hospitals and physicians	

- **Mixed:** Radioactive waste that also contains hazardous components regulated under RCRA.

- **Hazardous:** Under RCRA, waste that, because of its characteristics, may (1) cause or significantly contribute to an increase in mortality or an increase in serious, irreversible, or incapacitating reversible illness; or (2) pose a substantial present or potential hazard to human health or the environment when improperly treated, stored, transported, disposed of, or otherwise managed. Hazardous waste appears on special EPA lists or possesses at least one of the following characteristics: ignitability, corrosivity, reactivity, or toxicity. This category does not include source, special nuclear, or by-product material as defined by the Atomic Energy Act.

- **Nonhazardous:** Discarded material including solid, liquid, semisolid, or contained gaseous material resulting from industrial, commercial, mining, and agricultural operations and from community activities. This category does not include source, special nuclear, or by-product material as defined by the Atomic Energy Act.

Waste associated with the alternatives for treating the sodium-bonded spent nuclear fuel would be manag in existing or already-planned-for treatment, storage, and disposal facilities. The management of this wa could have an impact on existing site facilities. Waste generated during modifications to existing facilit could produce additional hazardous debris.

Waste management activities in support of treating sodium-bonded spent nuclear fuel would be contingent Records of Decision issued for the *Final Waste Management Programmatic Environmental Impact Stateme for Managing Treatment, Storage, and Disposal of Radioactive and Hazardous Waste* (DOE 1997. Depending on future waste type-specific Records of Decision, in accordance with that EIS, waste could treated and disposed of on site or at regionally or centrally located waste management centers. According the Transuranic Waste Record of Decision issued January 20, 1998, transuranic and transuranic mixed wa would be treated on site according to current planning-basis Waste Isolation Pilot Plant waste acceptan criteria and shipped to the Waste Isolation Pilot Plant for disposal. The impacts of disposing of transura waste at the Waste Isolation Pilot Plant are described in the *Waste Isolation Pilot Plant Disposal Phase Fir Supplemental Environmental Impact Statement* (DOE 1997b). Per the Hazardous Waste Record of Decisi issued August 5, 1998, nonwastewater hazardous waste would continue to be treated and disposed of at offs commercial facilities, with SRS continuing to treat some of its own hazardous waste on site in existi facilities, where this is economically favorable.

B.5.2 Description of Impact Assessment

As shown in **Table B–5**, impacts were assessed by comparing the projected waste stream volumes generat from the alternatives at each site with current site waste generation rates and storage volumes. For sodiu bonded spent nuclear fuel treatment, only the impacts related to the capacities of waste management facilit were considered. Environmental impacts of waste management facility operation are evaluated in otl facility-specific or site-wide National Environmental Policy Act (NEPA) documents. Projected wa generation rates for the alternatives were compared with the processing rates and capacities of those existi treatment, storage, and disposal facilities likely to be involved in managing the additional waste. Anotl factor considered is the reduction in volume of spent nuclear fuel and high-level radioactive waste destin for geologic disposal under each alternative.

The waste generation rates associated with sodium-bonded spent nuclear fuel treatment either were provic by the sites' technical personnel or were estimated based on evaluating similar processes, with adjustme made to account for differences in the amounts of materials processed.

B.6 Cumulative Impacts

Cumulative impacts can result from individually minor, but collectively significant actions taking place o a period of time (40 CFR 1508.7). The cumulative impact analysis for this EIS involved combining impacts of the sodium-bonded spent nuclear fuel treatment alternatives (including No Action) with the impa of other present and reasonably foreseeable activities in a region of influence.

The regions of influence for different resources can vary widely in extent. For example, the region of influe for waste management generally would be confined to the site itself; whereas the region of influence for hun health would include areas extending out to 80 kilometers (50 miles) from each site.

Table B–5 Impact Assessment Protocol for Waste Management

Resources	*Required Data*		*Measure of Impact*
	Affected Environment	*Alternative*	
Waste management capacity related to: - High-level radioactive waste - Transuranic waste - Low-level radioactive waste - Mixed waste - Hazardous waste - Nonhazardous waste	Site generation rates (cubic meters per year) for each waste type Site management capacities (cubic meters) or rates (cubic meters per year) for potentially affected treatment, storage, and disposal facilities for each waste type	Generation rates (cubic meters per year) of each waste type from modification and operation of existing facilities used to treat the sodium-bonded spent nuclear fuel	Combination of waste generation volumes from: (1) facilities that treat sodium-bonded spent nuclear fuel, and (2) current site and additional future generation volumes, in comparison to the capacities of applicable waste management facilities
Disposal capacity for transuranic waste (including mixed transuranic waste) *	Transuranic waste volume (cubic meters) expected to be disposed of at the Waste Isolation Pilot Plant Capacity at the Waste Isolation Pilot Plant (cubic meters)	Total transuranic waste generated (cubic meters) by spent nuclear fuel treatment facilities	Combination of transuranic waste generation volumes from: (1) facilities that treat sodium-bonded spent nuclear fuel, and (2) current site transuranic waste generation volume, in comparison to the capacity of the Waste Isolation Pilot Plant

* This additional entry is made for transuranic waste disposal because of its comparison with Waste Isolation Pilot Plant capacity.

In general, cumulative impacts were calculated by adding other planned and reasonably foreseeable future actions to the values for the baseline affected environment (i.e., conditions attributable to past and present actions by DOE and other public and private entities). This cumulative value was weighed against the appropriate impact indicators to determine the potential for impact. For this cumulative impact assessment, it was conservatively assumed that all facilities would operate concurrently at the DOE sites. Only selected indicators of cumulative impacts (**Table B–6**) were evaluated.

Table B–6 Selected Indicators of Cumulative Impacts

Category	*Indicator*
Resource use	Electricity use Water use Workers required
Air quality	Percent of NAAQS for criteria pollutants
Human health	Maximally exposed offsite individual, population, workers - dose - latent cancer fatalities
Waste	Site waste total and generation rate: - High-level radioactive waste - Transuranic waste - Low-level radioactive waste - Hazardous mixed waste

The analysis focused on the potential for cumulative impacts at each candidate site from DOE actions under detailed consideration at the time of this EIS (**Table B–7**). Non-DOE actions also were considered where information was readily available. Public documents prepared by agencies of Federal, state, and local governments were the primary sources of information for non-DOE actions.

Table B–7 Other Past, Present, and Reasonably Foreseeable Actions Included in the Cumulative Impact Assessments

Activities	*INEEL*	*SRS*
Surplus highly enriched uranium disposition		X
Surplus plutonium disposition		X
Interim management of nuclear materials at SRS		X
Management of waste	X	X
Radioactive releases from the Vogtle Nuclear Power Plant		X
Management of plutonium residues and scrub alloy at Rocky Flats		X
Construction and operation of a tritium extraction facility at SRS		X
Advanced mixed waste treatment project	X	
Defense waste processing facility		X
High-level waste and facility disposition	X	

It was assumed that construction impacts related to internal modification of existing facilities would not cumulative, because construction typically is short in duration and construction impacts generally temporary. Deactivation of the facilities utilized for the treatment of sodium-bonded spent nuclear fuel w not addressed in the cumulative impact estimates. Given the uncertainty regarding the timing of t deactivation and the fact that facilities could be used for other projects, any impact estimate at this time wou be premature. The evaluation of decontamination and decommissioning impacts will be provided in NEI documentation closer to the actual time of those actions.

Recent site-wide NEPA documents (**Table B–8**) provide the latest comprehensive evaluation of cumulat impacts for the sites.

Table B–8 Recent Comprehensive NEPA Documents for DOE Sites Assessed in This EIS

Site	*Document*	*Year*	*Record of Decisior First Issued*
INEEL	*Idaho High-Level Waste and Facilities Disposition Draft Environmental Impact Statement* (DOE 1999)	1999	—
SRS	*Savannah River Site Waste Management Final Environmental Impact Statement* (DOE 1995)	1995	October 1995

Appendix C

Technology Descriptions

APPENDIX C
TECHNOLOGY DESCRIPTIONS

The technology options that the U.S. Department of Energy (DOE) has considered for the treatment of sodium-bonded spent nuclear fuel are described in this appendix. Each technology is described in the context of treating sodium-bonded spent nuclear fuel driver and/or blanket assemblies. A brief discussion of the technical maturity of each treatment technology is included at the end of each technology description. The technical maturity of the technologies range from mature technologies that have been previously demonstrated by DOE for spent nuclear fuel or in an industrial setting to immature technologies that have only been demonstrated on a laboratory scale or for which only a conceptual design has been developed.

C.1 ELECTROMETALLURGICAL TREATMENT

The electrometallurgical treatment process for sodium-bonded spent nuclear fuel was developed at Argonne National Laboratory for processing Experimental Breeder Reactor-II (EBR-II) driver and blanket spent nuclear fuel assemblies. The process has been demonstrated for the stainless steel-clad uranium alloy fuel used in that reactor. The electrometallurgical treatment process uses electrorefining, an industrial technology used to produce pure metals from impure metal feedstock (DOE 1996). Although most of the sodium-bonded spent nuclear fuel driver and blanket elements are composed of uranium metal alloys, there also are small quantities (about 0.1 metric tons of heavy metal) of sodium-bonded uranium oxide, uranium nitride, and uranium carbide fuel. The oxide fuel would be prepared for treatment using the electrometallurgical treatment process by reducing the uranium oxide to uranium metal with lithium metal dissolved in small batches of lithium chloride-potassium chloride molten salt solution. The resulting uranium-bearing solution would be added to the molten salt solution used in the electrometallurgical treatment process for other sodium-bonded fuel and blanket elements and processed with those materials. The carbide fuel would be prepared for electrometallurgical treatment by cleaning the fuel of sodium to the extent possible and then converting the fuel to uranium oxide with water or dilute acid. This oxide then would be converted to uranium metal by lithium metal in a molten salt solution and processed by the electrometallurgical treatment process with other sodium-bonded spent nuclear fuel and blankets. The nitride fuel also would be prepared for electrometallurigcal treatment by converting it to uranium metal.

The description of electrometallurgical treatment in this environmental impact statement (EIS) is based on the assumption that the electrorefiner waste salts would be disposed of without salt recycling. This process differs from the original process described in the environmental assessment of the electrometallurgical treatment for the demonstration project (DOE 1996). In that assessment, the electrorefiner salts were to be treated in a series of zeolite columns. In these columns, the zeolite would absorb the fission products and transuranics from the salt and would release potassium as potassium chloride, which is one of the basic constituents of the electrorefiner salt. The bulk fluid handling system and zeolite columns were to be installed in the Hot Fuel Examination Facility argon cell. The potassium chloride salt and the recovered electrorefiner salts were to be reused in the electrorefiners. The fission products and transuranics that were absorbed in the zeolite then were to be removed from the columns in preparation for waste form production. Use of zeolite columns could potentially reduce the final ceramic waste volume.

Electrorefiner salt needs to be replaced if either the sodium concentration or the plutonium concentration limit is reached. The zeolite column would be a preferred option if the plutonium concentration in the salt became more limiting than the sodium concentration. In the latter case, which is the most likely scenario for the driver and Fermi-1 blanket spent nuclear fuel, the waste volume would be similar to the batch processing (i.e. without

salt recycling). During the demonstration project, due to lack of an available large-scale zeolite column, limited resources, and the fact that the batch processing produced acceptable waste forms and volumes, work focused primarily on batch processing. Additional research and development is needed to extend zeolite column use beyond the laboratory scale.

The individual steps in the electrometallurgical treatment process are described below. A diagram of the electrometallurgical process is shown in **Figure C–1**.

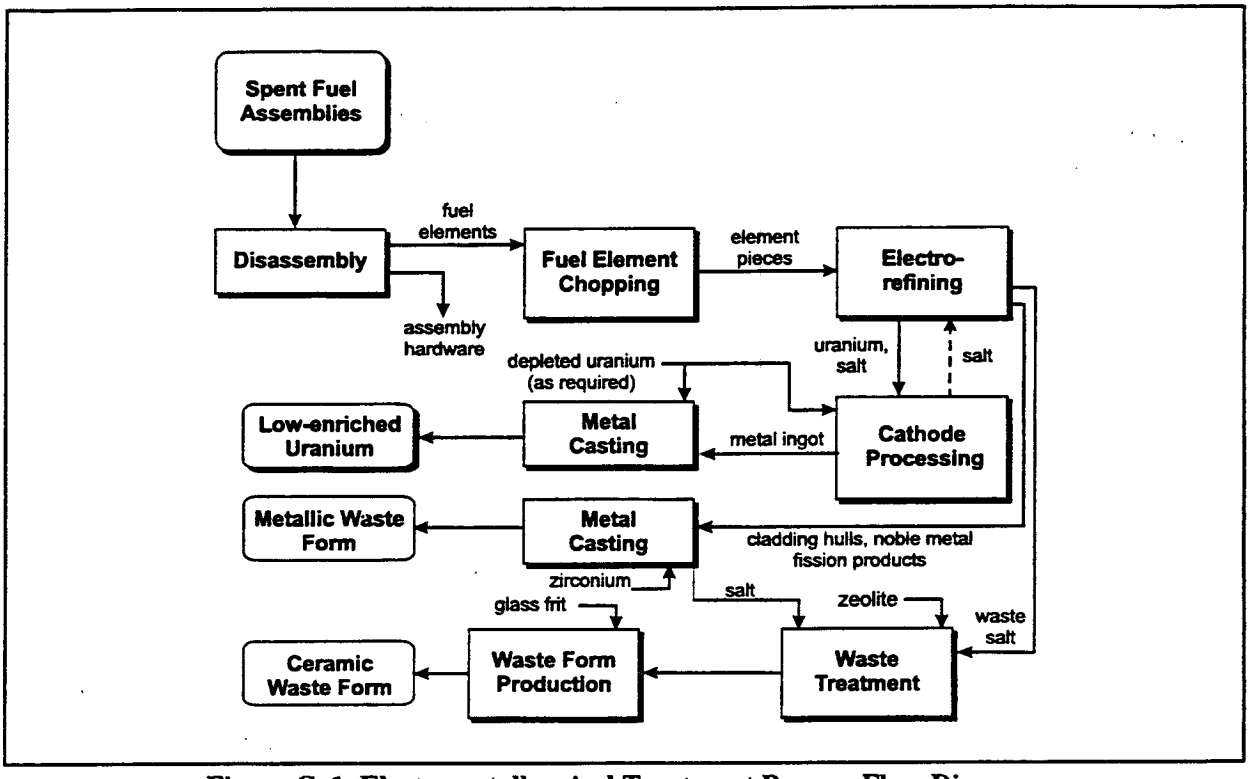

Figure C–1 Electrometallurgical Treatment Process Flow Diagram

Disassembly: Although the fuel and blanket assemblies mostly have been disassembled, some assemblies may need to be removed. The assembly hardware would be separated from the fuel elements that contain uranium and fission products by cutting the assemblies and physically separating the fuel elements. The fuel elements would be placed into a container for transfer to a hot cell containing an inert (argon) atmosphere for the remaining treatment steps. The assembly hardware would be stored at the Radioactive Scrap and Waste Facility at Argonne National Laboratory-West (ANL-W). This is a normal waste stream for ANL-W operations, and the separated hardware would be handled in accordance with normal site waste management practices.

Fuel Element Chopping: The sodium-bonded driver fuel or the blanket fuel elements would be placed in a machine for cutting into small pieces. The section of the element containing the fuel and sodium would be sheared into short segments. The section of the element containing the gas space (plenum) would be left intact. This section of the fuel pin cladding and the spacer wire would go into the metallic waste stream. The sheared fuel segments would be placed in perforated, stainless steel baskets to form an anode (positive electrode where oxidation would occur) for the electrorefiner.

During shearing of the hollow end (plenum) of the fuel pin, some fission product gases (primarily tritium and krypton) would be released to the argon cell atmosphere. These gases eventually would pass through high efficiency particulate air filters and be released up the emissions stack to the environment. All air emissions would be monitored and recorded.

Electrorefining: The electrorefiner is a machine in which the main electrometallurgical processes would occur. The electrorefiner vessel is made of steel. At its operating temperature of about 500 °C (930 °F), the vessel would contain a molten mixture of two salts, lithium chloride and potassium chloride. The electrorefiner also would have two or more electrodes: one or more anodes and one or more cathodes (negative electrodes where reduction would occur). Each anode would have baskets to hold the spent nuclear fuel pieces, and each cathode would consist of a bare steel rod where uranium metal would be collected.

The chopped fuel elements would be loaded into anode baskets and then lowered into the molten process salt. Upon application of an electric current between the anodes and cathodes, uranium, plutonium, and other transuranic elements, most of the fission products, and the sodium would be oxidized and dissolved into the salt. Uranium ions would be deposited at the cathode. Crystalline deposits of uranium would grow for 24 to 72 hours until almost all of the uranium in the anode baskets has been dissolved. The uranium-bearing cathodes would be raised into the gas space in the electrorefiner to allow some of the molten salt to drain away, although salt would adhere to each cathode. Each cathode then would be removed from the electrorefiner. The uranium deposit would be mechanically harvested and stored in the argon cell in a canister until it could be processed in the cathode processor.

The stainless steel cladding hulls and noble metal fission products would remain undissolved in the anode baskets. They would be removed from the electrorefiner and temporarily stored, prior to melting, into metallic waste-form ingots. The reactive fission products and transuranic elements would remain in the electrorefiner salt. The plutonium would be in a chloride compound in a liquid state and would be homogeneously mixed with other salts. The concentration of plutonium in the salt would be monitored through repeated sampling. The maximum plutonium concentration (about 8 weight percent) in the salt would not pose criticality safety concerns (Goff et al. 1999). In addition, abnormal localized concentrations of plutonium within the electrorefiner have been analyzed for a number of scenarios. These analyses have confirmed that an adequate margin of criticality safety would exist even under these conditions. The sodium would be in the form of sodium chloride (chemical form of table salt) as a part of the molten salt mixture.

The electrometallurgical process would use two electrorefining designs: Mark IV (for driver spent nuclear fuel) and Mark V (for blanket spent nuclear fuel). The Mark IV electrorefiner design would use a layer of cadmium to allow recovery of uranium that falls off the cathode during treatment. The Mark V design would use a collection basket instead of a cadmium layer.

Cathode Processing: The uranium deposits would be removed from the electrorefiner and treated to remove any adhering salt in the cathode processor, which is a furnace equipped with a vacuum system. The cathode product (along with depleted uranium to lower the enrichment of the resulting metallic ingot to less than 50 percent uranium-235, in the case of driver spent nuclear fuel) would be heated to about 1,200 °C (2,200 °F), melting both the uranium and the salt. Under vacuum conditions, the salt would distill away from the uranium and condense in a receiver crucible. The uranium would be melted in the cathode processor crucible and then solidified into an interim product ingot, which would be stored before final treatment in a casting furnace.

Uranium Metal Casting: The enriched uranium from driver spent nuclear fuel elements recovered in the electrorefiner would be melted together in a casting furnace with a separate stream of depleted uranium, electromagnetically stirred, and allowed to solidify. In this manner, enriched uranium from the treatment of driver spent nuclear fuel assemblies would be blended with depleted uranium in the casting furnace to form

low-enriched, metal ingots. The ingots then would be transferred to the Materials Storage Building within the Zero Power Physics Reactor complex, a controlled storage facility, until a decision is made by DOE regarding final disposition. Similarly, depleted uranium from treatment of blanket spent nuclear assemblies would be melted in a casting furnace and placed into storage until DOE makes a decision on final disposition.

Metallic Waste Form: The metallic waste form is one of the two high-level radioactive waste forms generated from electrometallurgical treatment of sodium-bonded spent nuclear fuel. This waste form would consist of metallic ingots used to stabilize the stainless-steel cladding material, residual fuel matrix materials, and noble metal fission products. Actinides that remain in the cladding hulls after dissolution also would be present in the metallic waste form. These metals would be melted together in a separate casting furnace from the one used for uranium metal casting. Any salt remaining with the metals would be distilled away under vacuum at about 1,200 °C (2,200 °F). Upon heating to about 1,500 °C (2,730 °F), the metals would melt and form an alloy. A small amount of zirconium metal also would be added to improve performance properties and to produce a lower melting point alloy. After cooling, the metal would solidify into a metallic waste ingot. The typical composition of these ingots would be stainless steel, 15 weight percent zirconium, and about 1 weight percent noble metal fission products (Goff et al. 1999). These ingots would be packaged and stored in interim dry storage at the Radioactive Scrap and Waste Facility until shipment to a geologic repository in canisters for disposal.

Treatment of Electrorefiner Waste: At the end of a processing campaign, fission products and actinides would remain dissolved in the molten salt. The waste salt would be removed from the electrorefiner and allowed to solidify. It then would be crushed and milled to obtain the desired particle size for ceramic waste form production. The liquid cadmium layer at the bottom of the electrorefiner also would be removed periodically, filtered, and returned to the electrorefiner. Filters from this bulk fluid handling system would become part of the metallic waste stream.

Ceramic Waste Form Production: The ceramic waste form is the second waste form generated from electrometallurgical treatment of sodium-bonded spent nuclear fuel. The crushed and milled waste salt and dried zeolite would be added to a heated V-mixer. (Zeolites are crystalline aluminosilicates of group I (alkali) and group II (alkaline earth) elements. Their framework is a network of aluminum oxide and silicon oxide tetrahedra linked by the sharing of oxygen atoms. The networks of tetrahedra in the zeolite form cages in which molecules can be occluded.) The waste salt containing fission products and actinides would be absorbed into the crystal lattice of the zeolite, forming a dry particulate solid. Glass frit (sand-like glass) then would be mixed with the waste-bearing zeolite and placed in a special metal canister designed to be compressed to a desired and predictable shape. The mixture of material going into the process would be about 75 weight-percent waste-bearing zeolite and 25 weight-percent glass (Goff et al. 1999). This canister would be put into a type of furnace called a hot isostatic press, where it would be subjected to a temperature of 850 °C (1,560 °F) and a pressure of 1,057 kilograms per square centimeter (15,000 pounds per square inch). This would compress the canister and transform the material inside into a single cylinder of glass-bonded zeolite, which is referred to as the ceramic waste form. During compression, the zeolite would be converted to sodalite, a naturally occurring, salt-bearing material. Fission product chlorides largely would remain in the sodalite phase, while actinides (and most of the rare earth elements) would react with residual water in the zeolite to form oxide phases in the waste form. A conservative criticality assessment of the ceramic waste form indicated that the plutonium concentration in the waste form would pose no criticality safety concern (ANL 1999). These waste-form cylinders would be packaged and stored in interim dry storage at the Radioactive Scrap and Waste Facility until shipment to a geologic repository in canisters for disposal.

Technology Maturity: The electrometallurgical treatment process is considered to be a mature technology. DOE demonstrated the electrometallurgical process for stainless steel-clad uranium alloy fuel used in the EBR-II reactor. Furthermore, it is an industrial technology used to produce pure metals from impure metal feedstock.

C.2 DECLAD AND CLEAN PROCESS

Cleaning (removing metallic sodium) and/or decladding are necessary steps in the treatment of sodium-bonded spent nuclear fuel using the plutonium-uranium extraction (PUREX) process at the Savannah River Site (SRS), the melt and dilute process at either SRS or ANL-W, and high-integrity can packaging. The fuel would need only to be cleaned of metallic sodium (i.e., it would not have to be declad) for melt and dilute processing at ANL-W and high-integrity can packaging. Decladding and sodium removal could be done using either a mechanical process (the melt, drain, evaporate, and calcine [MEDEC] process) or a laser declad and alcohol wash process. In the MEDEC process, the metallic sodium would be removed first; then, if necessary, the fuel would be declad. The MEDEC process has been performed for unirradiated blanket elements. The laser declad and clean process performs these functions in the reverse order—the fuel is declad using the laser and then the sodium is removed using an alcohol wash. Laser cutting is accompanied by partial volatilization of cladding, sodium, and materials dissolved or suspended in sodium, most notably cesium. This process was performed at the Rockwell International Hot Laboratory to declad and remove metallic sodium from approximately 7,000 EBR-II irradiated (low burnup) blanket spent nuclear fuel elements (Frazier and Campbell 1987). The process used a yttrium-aluminum-garnet (YAG) laser system. An automated fuel cutting sequence was developed, and the cladding was cut into strips. The cutting sequence included four circumferential cuts and three longitudinal cuts (at 120° circumferential segments) to allow mechanical removal of the cladding pieces. The bare fuel pins and the cladding pieces were washed in alcohol (ethanol) and water mixtures to remove the metallic sodium. The bare fuel pins were packaged and sent to SRS for processing. The contaminated alcohol mixture then was evaporated to reduce the volume, solidified with a grouting agent, and disposed of as low-level radioactive waste.

MEDEC Sodium Removal and Processing: This process would be performed at the Hot Fuel Examination Facility. Fuel elements would be brought into an argon-atmosphere hot cell where the ends of the elements would be cut off to expose the sodium within the cladding. The elements then would be cut into segments less than 61 centimeters (24 inches) in length. The fuel elements would be placed into a crucible and loaded into a closed induction furnace with an off-gas control system. The temperature in the furnace would be raised above the melting point of sodium (about 200°C [390°F]) and the molten sodium would be drained into a collection tank. With most of the sodium removed, the temperature would be raised to about 500°C (930°F) and a 10^{-4} torr vacuum would be applied to the chamber. This vacuum would volatilize the residual sodium, allowing the sodium vapor to be drawn away from the fuel. The vapor-phase sodium would be condensed in a trap and combined with the drained sodium in the collection tank pending further processing. Operating conditions necessary for complete sodium removal would be determined through testing. Verification of sodium removal would be obtained through analytical laboratory inspection and analysis. If the MEDEC process is applied for production, further verification would be performed. If necessary, the fuel pins would be mechanically pushed out of the stainless steel cladding after all the metallic sodium is eliminated.

Sodium recovered during the cleaning process would contain some fission products, most notably cesium-137. This cesium would be recovered by vacuum distillation of the sodium, taking advantage of the large difference in the boiling points of the two elements. The boiling point of cesium is 690°C (1,274°F), while the boiling point of sodium is 892°C (1,638°F). A vapor trap would be placed between the distillation column and pump to collect volatile species emitted from the condenser. The purified sodium would be processed by injection into a chamber where it would react rapidly with oxygen and water to form an aqueous sodium hydroxide. Carbon dioxide gas then would be bubbled through the hydroxide solution, converting the sodium hydroxide to a sodium carbonate. The aqueous sodium carbonate would be solidified with a binder and packaged for disposal as low-level radioactive waste. The cesium collected as distillate from the separation process would be added to the ceramic waste form described in Section C.1.

Laser Declad and Wash: As stated above, this process was performed at Rockwell International. The process would use a modified laser system for remote operations and a machine to hold and index the fuel elements

during cutting operations. The fuel elements would be brought into a hot cell. The fuel would be cut both circumferentially and longitudinally in a predefined cutting sequence. The fumes generated during the cutting process would be filtered and exhausted through an off-gas system. The fuel pins along with the cladding strips would be washed in an alcohol and water mixture to neutralize the metallic sodium and fission product (i.e., cesium) contamination. The fuel pins would be packaged and stored at ANL-W for further treatment, or sent to SRS for treatment. The alcohol and water solution would be evaporated partially, and the sodium/cesium alcoholates and hydroxides would be neutralized, then solidified in a grouting agent and disposed of as a low- or high-level radioactive waste, depending on the cesium content.

Compatibility with argon cell operation: The MEDEC procedure has been demonstrated using sealed vessels in an ANL-W facility. The laser process was demonstrated at Rockwell International. The laser process operation required personnel to enter the hot cell on a biweekly basis for laser maintenance and purging of the cell atmosphere to maintain a low oxygen level (less than 4 percent) and to vent alcohol-water vapors and hydrogen gas from the cell. Neither of these practices would be acceptable for argon cell operations. These concerns could be ameliorated by use of a hot cell with a different type of inert atmosphere, such as nitrogen, which could be purged and replaced with air. Nevertheless, personnel entry into the hot cell would still be restricted due to current radiation exposure controls and the higher level of fission products in the present inventory of sodium-bonded spent nuclear fuel. The use of alcohol and water in a multipurpose hot cell could raise a criticality concern if fissile materials were present in the cell. However, there would be no criticality issue with the blanket fuel itself, so a dedicated hot cell would eliminate this concern.

Finally, sodium collected during previous laser decladding operations was disposed of as low- level radioactive waste. The sodium collected from the processing of the fuel addressed by this EIS would be contaminated with cesium. If sufficient quantities of cesium were present in the sodium, this waste could not be treated as low-level radioactive waste. For the sodium to be managed as low-level radioactive waste, the sodium would have to be processed (as is done with the sodium removed from the fuel in the MEDEC process) to remove the cesium from the alcohol mixture using a currently undefined process. Because of the compatibility concerns associated with laser operation in the argon cell, the MEDEC process has been used to evaluate the various alternatives that require cleaning and/or decladding of the sodium-bonded spent nuclear fuel.

Technology Maturity: Argonne National Laboratory has used the MEDEC process to recover the uranium from 1,700 unirradiated sodium-bonded fuel rods. Laser decladding and cleaning were demonstrated on 17 metric tons of heavy metal (6,780 rods) of very low burnup EBR-II blanket fuel. Both processes are considered mature technologies.

C.3 PUREX PROCESS

The PUREX process is a counter-current solvent extraction method used to separate and purify uranium and plutonium from fission product-containing spent nuclear fuel and irradiated uranium targets. DOE has two facilities at the SRS, F-Canyon and H-Canyon, that use the PUREX process for the treatment of aluminum-clad fuel and targets. In this EIS, the PUREX process at F-Canyon is being considered for treating declad and cleaned EBR-II and Fermi-1 blanket spent nuclear fuel. The stainless steel cladding and sodium would be removed from these blanket spent nuclear fuel elements at ANL-W. The cleaned blanket spent nuclear fuel pins would be packaged in aluminum cans and shipped to SRS. The decladding and cleaning activities would be conducted in argon cells at ANL-W facilities. A diagram of the PUREX process is shown in **Figure C–2.**

Disassembly: The first step in the process would be similar to the disassembly process previously described in Section C.1. The assembly hardware would be stored at the Radioactive Scrap and Waste Facility at ANL-W and handled in accordance with normal site waste management practices.

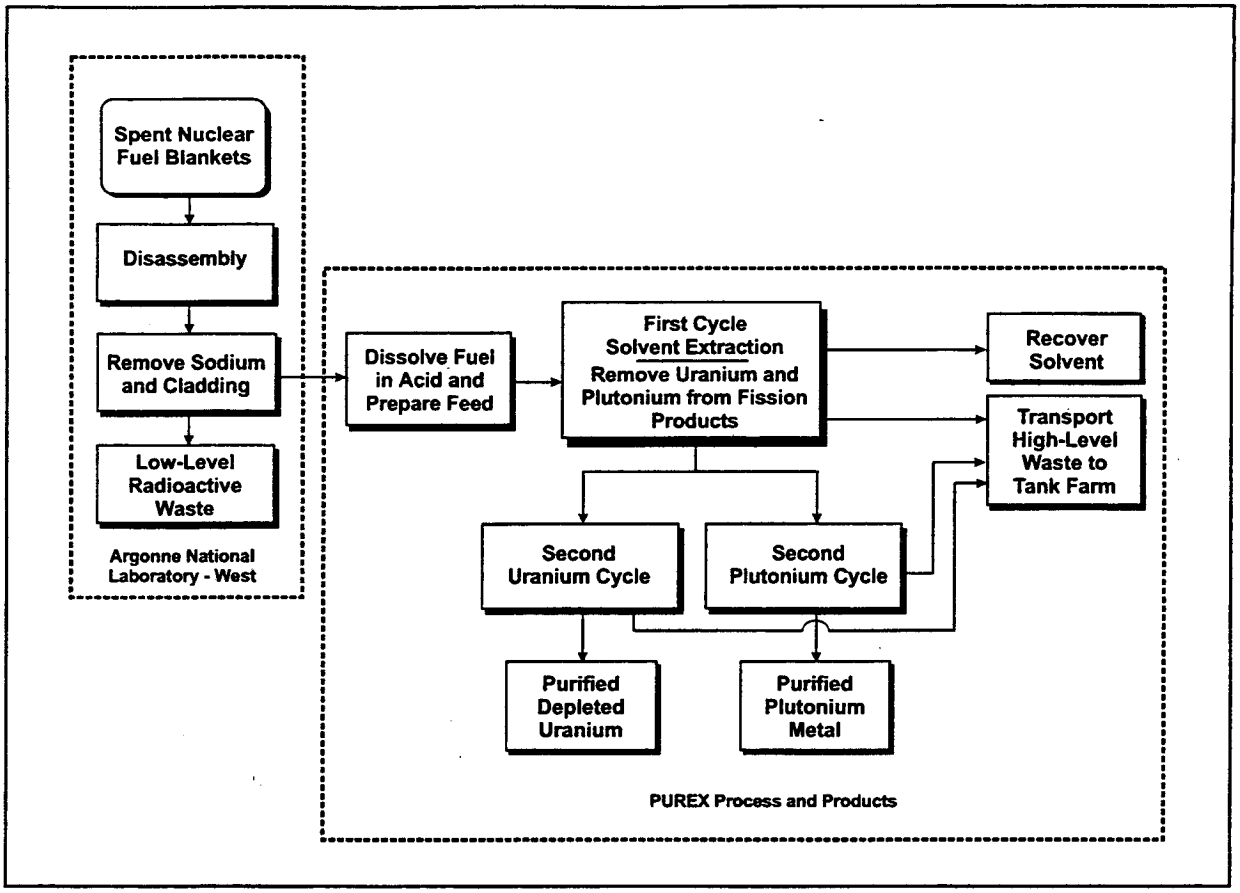

Figure C–2 PUREX Process Flow Diagram at SRS

Decladding and Sodium Removal: Blanket spent nuclear fuel elements would be cleaned and declad using the MEDEC process described in Section C.2. The uranium pins would be mechanically pushed out of the stainless steel cladding after all the metallic sodium has been eliminated. The bare uranium pins then would be packed into aluminum canisters in the Hot Fuel Examination Facility. The canisters, approximately 10 centimeters (4 inches) in diameter and 61 centimeters (24 inches) in length, would be backfilled with an inert gas and sealed. Each canister would contain about 60 kilograms (130 pounds) of depleted uranium fuel pins. The canisters would be placed in a NAC-LWT cask for shipment to SRS.

Receiving and Storage at SRS: The packages of blanket spent nuclear fuel pins from ANL-W would be received at the L-Reactor Disassembly Basin for storage until transfer to the F-Canyon for stabilization using the PUREX process.

PUREX Unit Operations: The EBR-II and Fermi-1 blanket spent nuclear fuel pins would be processed using the traditional PUREX process. This process consists of several major operations referred to as "unit operations," which yield two products, uranium and plutonium (in solution form). The unit operations are dissolution, head end, first cycle, second uranium cycle, and second plutonium cycle. Unit operations that support the product recovery process are high-activity waste, low-activity waste, and solvent recovery.

Dissolution and Head End: The blanket fuel would be transferred to the canyon in casks and loaded into a large tank called a dissolver. Heated nitric acid in the tank would dissolve the blanket fuel, resulting in a solution containing depleted uranium, plutonium, and fission products. Gelatin would be added to the solution, if necessary, to precipitate fuel impurities. Then the solution would be transferred to a centrifuge

where impurities would be removed as waste. The clarified product solution from this process would be adjusted with nitric acid and water in preparation for the first cycle unit operation in the PUREX process. The waste stream generated from the process would be chemically neutralized and sent to the SRS high-level radioactive waste tanks pending further processing at the Defense Waste Processing Facility.

First-Cycle Operation: The first-cycle operation has two functions: (1) to remove fission products and other chemical impurities, and (2) to separate the solution into two product streams (i.e., uranium and plutonium) for further processing. This separation process occurs as the product solution passes through a series of equipment consisting of a centrifugal contactor and mixer-settler banks. Before the introduction of the product solution, flows of solvent and acid solution would be started through the equipment. After an equilibrium condition has been established, the product solution would be introduced. The chemical properties of the acid/solvent/product solutions in contact with each other would cause the fission products to separate from the uranium and plutonium. Later in the first cycle process, the plutonium would be separated from the uranium in a similar manner. The first cycle would produce four process streams: (1) a plutonium-containing solution (with some residual fission products), which would be sent to the second plutonium cycle; (2) a uranium-containing solution (with some residual fission products), which would be sent to the second uranium cycle; (3) a solvent stream, which would be sent to a solvent recovery cycle; and (4) an aqueous acid stream, which would contain most of the fission products and would be sent to the SRS high-level radioactive waste tanks pending further processing at the Defense Waste Processing Facility.

Second Uranium Cycle: In the second uranium cycle, the uranium-containing solution coming from the first cycle would be purified further in a manner similar to that described for the first cycle. The purified solution would be transferred to storage tanks. Eventually, the uranium would be converted to uranium oxide and stored in 208-liter (55-gallon) drums. The uranium oxide would be stored for future use. The solution containing the residual fission products would be sent to SRS high-level radioactive waste tanks pending further processing at the Defense Waste Processing Facility.

Second Plutonium Cycle: In the second plutonium cycle, the plutonium-containing solution coming from the first cycle would be further purified in a manner similar to that described for the first cycle. The purified solution would be converted to plutonium metal in the FB-Line. The plutonium would be disposed of in accordance with the Record of Decision (75 FR 1608), for the *Surplus Plutonium Disposition Final Environmental Impact Statement* (DOE 1999). The solution containing the residual fission products would be sent to the SRS high-level radioactive waste tanks pending further processing at the Defense Waste Processing Facility.

Other Unit Operations: The unit operations for high- and low-activity waste would reduce the volume of the aqueous streams containing fission products. The streams originate with primary separation process unit operations such as the first cycle. The fission products would be separated and sent to the high-level radioactive waste tanks. The volume reduction process would be accomplished using a series of evaporators in the canyons. The solvent recovery unit operation would recover and recycle the solvent used in the first cycle by removing impurities from the solvent. The purified solvent would be returned to the first cycle for reuse, and the impurities would be transferred to low-activity waste for processing (DOE 1994).

Technology Maturity: The PUREX process is considered to be a mature technology. It has been used throughout the world since 1954 to separate and purify uranium and plutonium from fission product-containing spent nuclear fuel and irradiated uranium targets.

C.4 HIGH-INTEGRITY CAN PACKAGING

The high-integrity can packaging option is being considered for EBR-II and Fermi-1 blanket spent nuclear fuel elements. The high-integrity can is made from Hastelloy Alloy C-22 metal alloy pipe having a 13.7-centimeter (5-inch) diameter, variable length, and a pipe wall thickness of 0.655 centimeters (0.258 inches) (Shaber 1998). Hastelloy Alloy C-22 is an alloy of nickel, chromium, and molybdenum that is highly corrosion-resistant due to its high chrominium (22 percent) and molybdenum (13 percent) content. The high-integrity cans are designed for dry hot cell loading with a lid adaptable to wet loading and vacuum drying. The lid on each can has a threaded design to accommodate the partial loading of the spent nuclear fuel into the can at different times. The threaded lid prevents spillage of can contents during interim storage (DOE 1998). After packaging the fuel, the cans would be placed in standardized canisters of about 46 to 61 centimeters (18 to 24 inches) in diameter and up to 3 meters (118 inches) in length and would be codisposed with high-level radioactive waste in a repository. A diagram of the high-integrity can packaging is shown in **Figure C–3**.

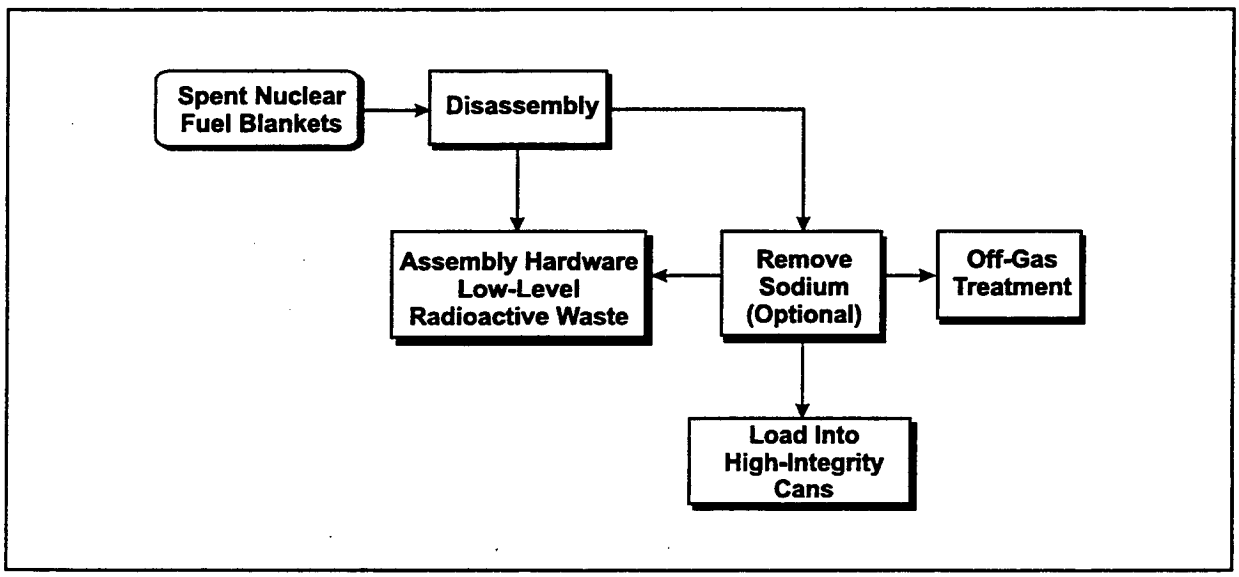

Figure C–3 High-Integrity Can Packaging Flow Diagram

Disassembly: Although the blanket assemblies have been mostly disassembled, there may be some assembly hardware that needs to be removed. The assembly hardware would be separated from the blanket fuel pins by cutting the assemblies and physically separating the fuel elements. The fuel elements would be placed into a container for transfer to an argon-atmosphere hot cell for the remaining process steps. The assembly hardware would be stored at the Radioactive Scrap and Waste Facility at ANL-W. This is a normal waste stream for ANL-W operations, and the separated hardware would be handled in accordance with normal site waste management practices.

Sodium Removal: If needed, the blanket spent nuclear fuel elements would be cleaned using the MEDEC process described in Section C.2.

Loading into High-Integrity Cans: The blanket spent nuclear fuel elements would be packaged in a standard-sized can fabricated from Hastelloy Alloy C-22, or possibly some other highly corrosion-resistant materials such as titanium Grade-12.

The high-integrity can would be placed in dry storage at an appropriate location. If transportation is required, the cans would be packaged into shipping casks. Prior to shipment to a geologic repository, the high-integrity

can containing spent nuclear fuel would be placed into a standardized canister, an overpack designed to provide additional containment within the waste package under repository conditions.

Direct Disposal of Sodium-Bonded Spent Nuclear Fuel: Direct disposal of sodium-bonded spent nuclear fuel is currently precluded by DOE policy concerning acceptance of Resource Conservation and Recovery Act-designated mixed waste (which contains both hazardous and radioactive waste). In the absence of such a policy, sodium-bonded spent nuclear fuel (driver and blanket) could be cleaned of surface sodium, packaged in high-integrity cans without removal of metallic sodium from the interior of the fuel elements, and directly disposed of in a geologic repository. The high-integrity cans would be placed into a standardized canister designed to promote containment under repository conditions.

Technology Maturity: Packaging materials in a high-integrity can is considered to be a mature technology. These cans would be made from highly corrosion-resistant materials and would be designed to provide exceptional protection from external environments.

C.5 Melt and Dilute Process

The melt and dilute process is being considered for driver and blanket spent nuclear fuel elements. Three process options are being considered: (1) melting bare uranium blanket spent nuclear fuel pins with aluminum, (2) melting blanket spent nuclear fuel elements with cladding and additional stainless steel, and (3) using a modified melt and dilute process capable of handling the sodium in a volatilized form and processing chopped driver spent nuclear fuel elements that could not be completely cleaned of sodium. Processing activities would be conducted in the Hot Fuel Examination Facility at ANL-W or in Building 105-L at SRS. A diagram of the melt and dilute process flow for the first two options is shown in **Figure C–4**. A process flow diagram for the third option is shown in **Figure C–5**.

Disassembly of Blanket Spent Nuclear Fuel Elements at ANL-W: Although the blanket spent nuclear fuel assemblies mostly have been disassembled, there may be some assembly hardware that needs to be removed. The assembly hardware would be separated from the blanket spent nuclear fuel elements by cutting the assemblies and physically separating the fuel elements. The spent nuclear fuel elements would be placed into a container for transfer to an argon-atmosphere hot cell for the remaining process steps. The assembly hardware would be stored at the Radioactive Scrap and Waste Facility at ANL-W. This is a normal waste stream for ANL-W operations, and the separated hardware would be handled in accordance with normal site waste management practices.

Sodium Removal and Processing at ANL-W: Blanket spent nuclear fuel elements would be cleaned using the MEDEC process described in Section C.2.

Decladding and Packaging Blanket Spent Nuclear Fuel Pins for Shipment to SRS: In the first melt and dilute processing option, blanket spent nuclear fuel pins that would be sent to SRS would be mechanically pushed out of the stainless steel cladding after all the sodium has been removed. These blanket spent nuclear fuel pins would be packed into aluminum cans in the Hot Fuel Examination Facility. The cans, approximately 10 centimeters (4 inches) in diameter and 61 centimeters (24 inches) in length, would be backfilled with an inert gas and sealed. Each can would contain about 60 kilograms (130 pounds) of depleted uranium spent nuclear fuel pins. The cans would be packaged and placed in a NAC-LWT cask for shipment to SRS.

Receiving and Storage at SRS: The cleaned and declad blanket spent nuclear fuel cans from ANL-W would be received at the L-Reactor Disassembly Basin for storage until transfer to the processing facility in Building 105-L.

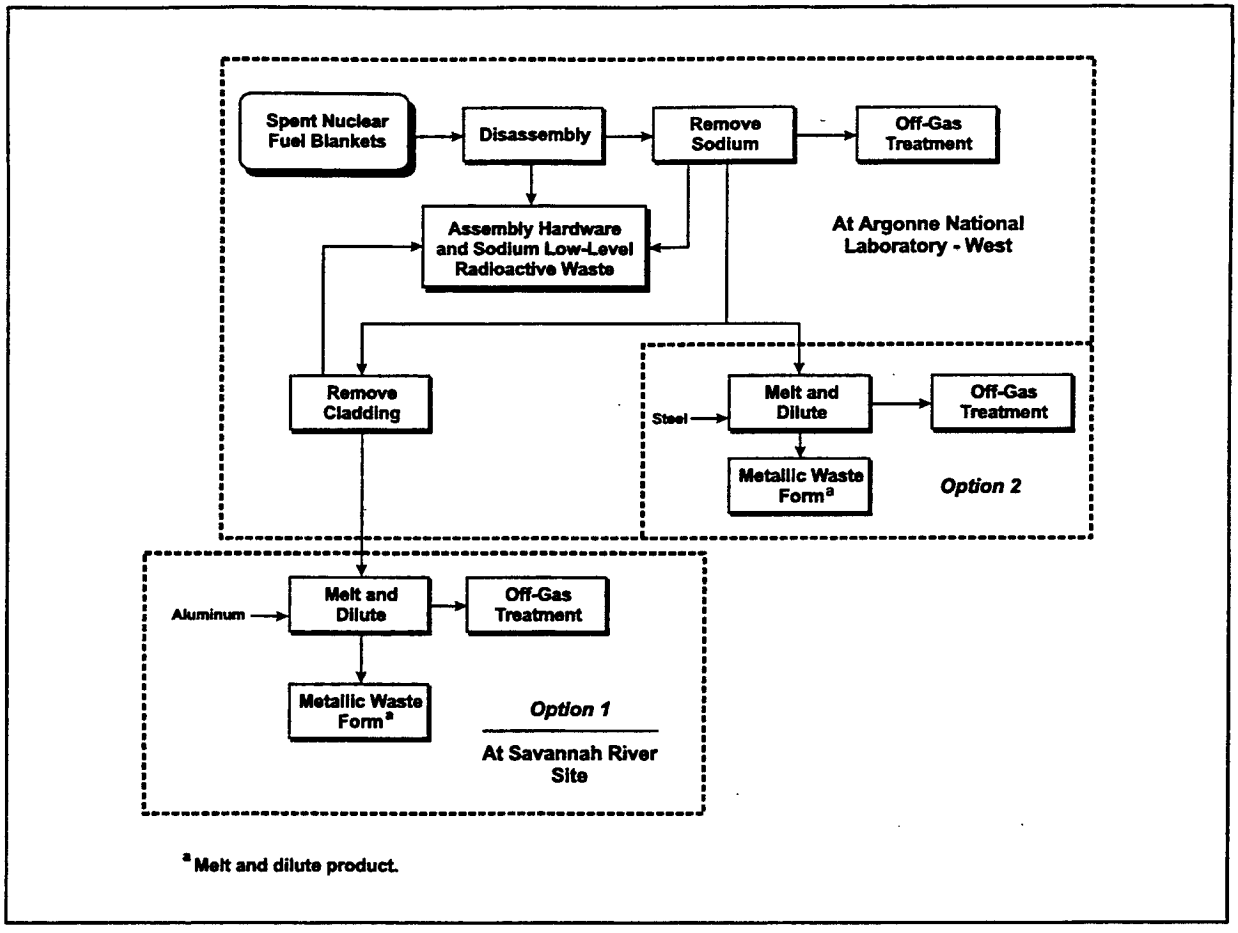

Figure C–4 Melt and Dilute Process Flow Diagram (Options 1 and 2)

Melt and Dilute Process for Blanket Fuel at SRS: Blanket spent nuclear fuel cans would be transferred to the treatment facility in Building 105-L for processing. The spent fuel cans would be loaded into an induction furnace where they would be heated to approximately 1,000 °C (1,830 °F). This temperature significantly exceeds the aluminum-uranium eutectic temperature required to initiate the melting, so it would proceed within a reasonable time. Sufficient aluminum would be added to make an aluminum-uranium alloy with a composition of about 70 percent aluminum and 30 percent uranium. The metal alloy would be cast into an ingot, sampled, and packaged into canisters. The canisters would be evacuated, filled with inert gas, sealed by welding, and transferred to storage pending disposition in a geologic repository. Volatile fission products would be captured by a series of filter banks before releasing the off-gas. The filters would be disposed of as low-level or high-level radioactive waste, as appropriate.

Melt and Dilute Process for Blanket Fuel at ANL-W: In the second melt and dilute processing option, blanket spent nuclear fuel elements recovered from the sodium removal process would be placed in an induction furnace crucible with additional radioactive waste steel. Sufficient steel would be added to make an alloy with a composition of about 50 percent each of uranium and steel. The furnace would be heated to approximately 1,400° C (2,550° F) to melt the uranium, after which the steel would be slowly dissolved into the uranium pool. The mixture would be electromagnetically stirred to a uniform composition. The metal alloy would be cast into an ingot, sampled, and packaged for interim storage at the Radioactive Scrap and Waste Facility. An off-gas system would capture the volatile and semi-volatile fission products for stabilization and processing into waste forms suitable for disposal. The filters would be disposed of as low-level or high-level radioactive waste, as appropriate.

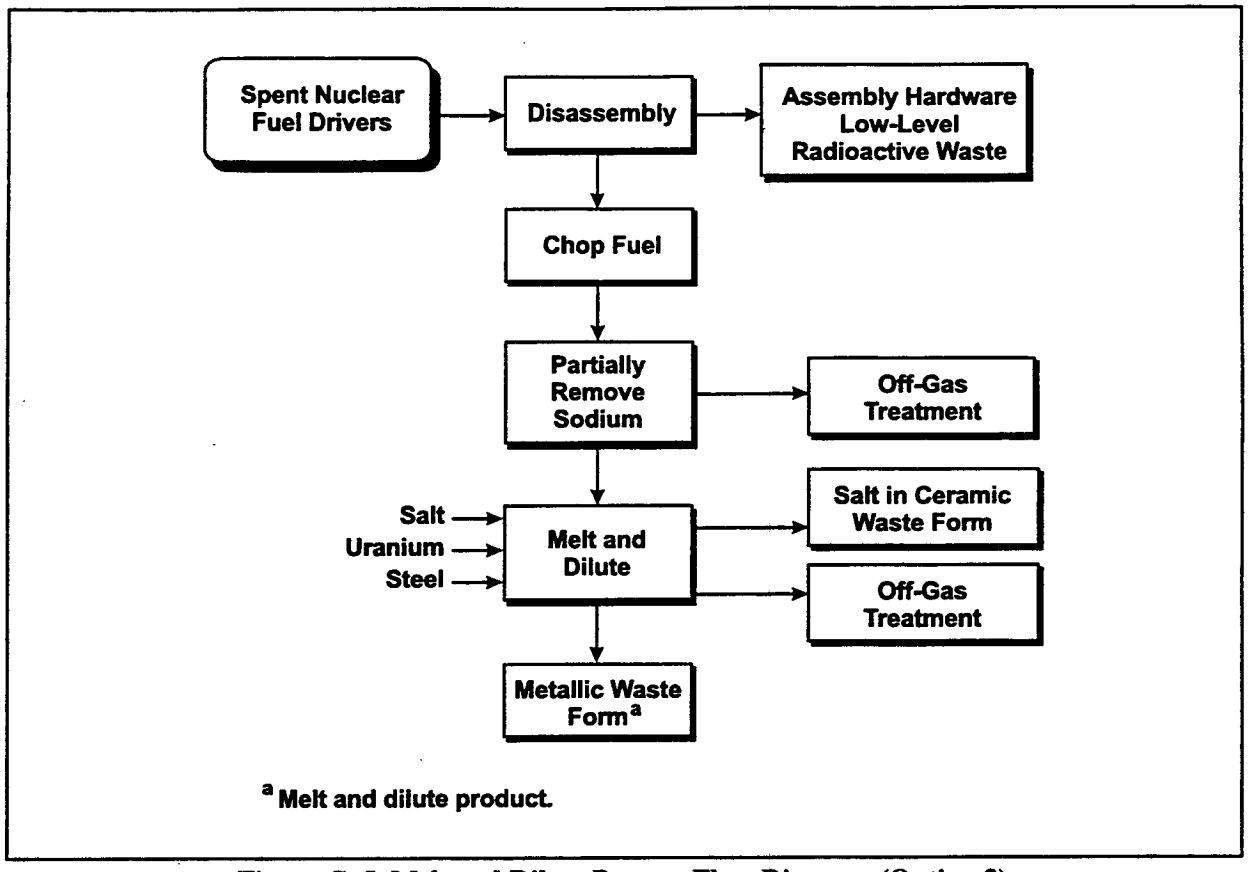

Figure C–5 Melt and Dilute Process Flow Diagram (Option 3)

Melt and Dilute Process for Driver Fuel at ANL-W: The third melt and dilute option would be for sodium-bonded driver spent nuclear fuel. Most of the metallic sodium in the driver spent nuclear fuel elements would be removed in a manner similar to the way sodium would be removed from the blanket spent nuclear fuel elements, i.e., the fuel would be cut into a few segments and heated to allow some sodium to drain away and then the fuel pieces would be heated under a vacuum to volatilize additional sodium. However, as explained below, not all the sodium could be removed by these processes. During the irradiation of the fuel in the reactor, after approximately a 1 to 3 percent burnup, the gap between the fuel pin and the cladding would be closed by the swelling of the fuel pin and interdiffusion between the cladding and the fuel pin. During the swelling process, the fuel pin would become porous and metallic sodium would enter the fuel. At discharge from the reactor, 15 to 20 percent of the fuel's pores would contain trapped sodium with dissolved fission products. The trapped sodium within the fuel pins and the areas of interdiffused fuel and cladding could not be removed.

Since not all the sodium could be removed from the driver spent nuclear fuel by the heating and vacuum process, a modified melt and dilute process would be needed. In this process, the driver fuel elements would be covered with a layer of low melting-temperature salt containing uranium chloride to oxidize the molten sodium. Depleted uranium would be added in a ratio of about 2.5 to 1 to reduce the enrichment to less than 20 percent uranium-235. Radioactive waste steel would be added in equal weight to the uranium to complete the mix. The furnace then would be heated to a temperature of about 1,400 °C (2,550 °F). The molten salt would capture sodium vapors escaping from the fuel elements as they melt, protecting the downstream components from the sodium. After volatilization of the sodium and reaction with the molten salt, a vacuum would be applied to the furnace to volatilize the salt, which would be condensed and partially reused. The salt

would be stabilized in the ceramic waste form described in Section C.1. The molten metal would be stirred to achieve a uniform composition, and then would be cast into an ingot, placed into a container, and stored. An off-gas system would capture the volatile and semi-volatile fission products for stabilization and processing into waste forms suitable for disposal. The filters would be disposed of as low-level or high-level radioactive waste, as appropriate.

Technical Maturity: The melt and dilute process was developed for treating aluminum-based spent nuclear fuel at SRS and is DOE's preferred technology for treating most (almost 97 percent by volume) of that type of spent nuclear fuel (DOE 2000). The melt and dilute process for stainless steel-clad spent nuclear fuel would require operating temperatures of approximately 1,400 °C (2,550 °F), compared with about 1,000 °C (1,830 °F) for aluminum-based spent nuclear fuel. Induction-heated melters that can achieve the higher temperatures required for stainless steel have been demonstrated at ANL-W. Technology development would be required to demonstrate capturing the quantities of sodium present in the driver spent nuclear fuel assemblies in a molten salt.

The melt and dilute process can be used for most of the driver spent nuclear fuel. However, there are small quantities (about 0.1 metric tons of heavy metal) of driver spent nuclear fuel that are composed of uranium oxide, uranium carbide or uranium/plutonium carbide, and uranium nitride, which have high melting points and cannot be treated using the melt dilute process.

C.6 Direct Plasma Arc-Vitreous Ceramic Treatment Process

The plasma arc treatment technology (DOE 2000) would use a plasma torch to melt and oxidize the spent nuclear fuel in conjunction with depleted uranium oxide and other ceramic-forming materials, as necessary. The fuel would be fed into the process with minimal sizing or pretreatment. The plasma arc would cut the fuel assemblies into small pieces and heat the fuel to temperatures at least as high as 1,600 °C (2,910 °F) to melt and oxidize it in a rotating furnace. Ceramic material would be added, as necessary, while the mixture was being homogenized by the torch. When melting and oxidation were complete, the rotating furnace would slow and the melt would fall into molds prepared to receive it. A diagram of the plasma arc treatment process flow is shown in **Figure C–6**.

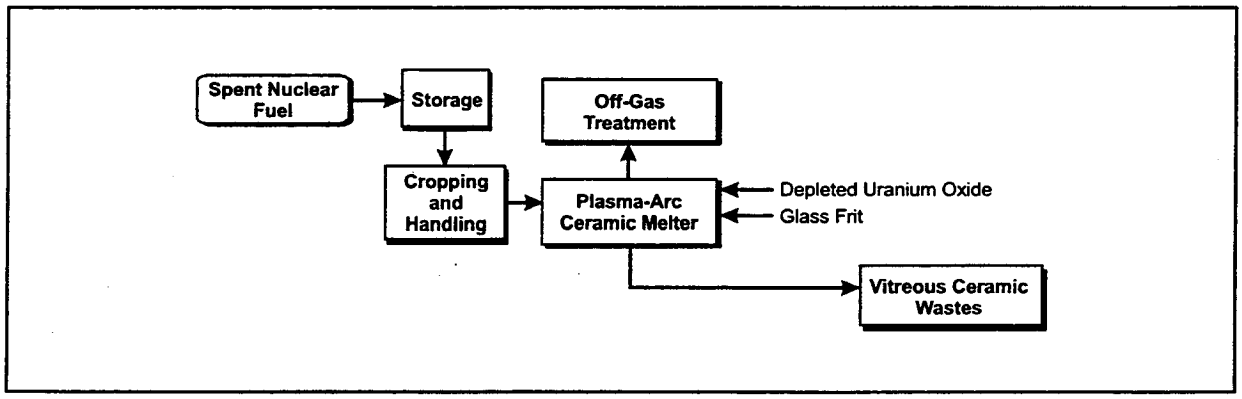

Figure C–6 Direct Plasma Arc-Vitreous Ceramic Treatment Process Flow Diagram

Metallic fuel such as EBR-II fuel would require the addition of some ceramic material. Depleted uranium could be added to the process in almost any form to reduce the uranium-235 enrichment. Criticality issues would be addressed by limiting the process to batch runs of preselected quantities of fissile material by the addition of the depleted uranium and by the addition of neutron poisons, if necessary.

As with all processes that dissolve or melt spent nuclear fuel, the plasma arc treatment would produce radioactive off-gases. These gases would be filtered and treated by appropriate means, with the filter and treatment media recycled into the plasma arc furnace for incorporation into the ceramic product.

Technology Maturity: The plasma arc process is a developmental technology that has not been demonstrated for stabilization of spent nuclear fuel.

C.7 Glass Material Oxidation and Dissolution System

The Glass Material Oxidation and Dissolution System (GMODS) uses lead oxide to convert unprocessed spent nuclear fuel directly to borosilicate glass using a batch process. A diagram of the GMODS process flow is shown in **Figure C–7.**

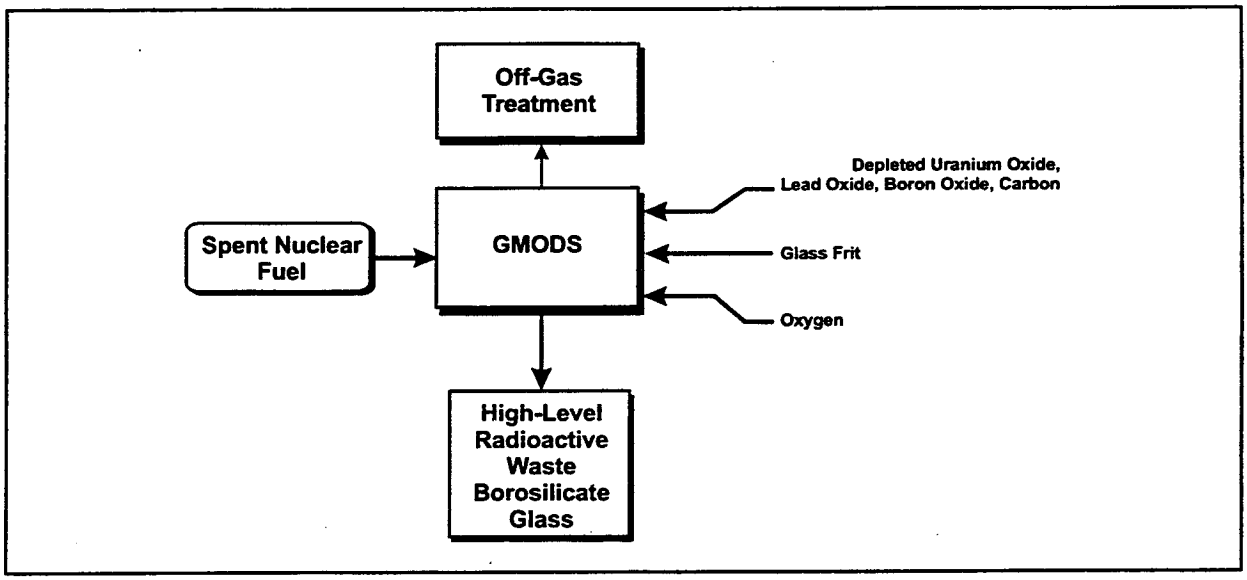

Figure C–7 GMODS Process Flow Diagram

Metal Oxidation: The principal piece of equipment for GMODS would be an induction-heated, cold-wall melter, which is used commercially to convert corrosive or high-melting metals to ultrapure materials. The melter, operating at 1,000 to 1,200 °C (1,830 to 2,200 °F), would be charged with a molten glass consisting of lead oxide and boron oxide. Oxides and amorphous components of the spent nuclear fuel would directly dissolve into the glass. Metals, which normally do not dissolve in glass, would be converted to oxides by the lead oxide. Boron oxide, a neutron poison, is a common agent for dissolving oxides into glass. Criticality concerns would be addressed by diluting the uranium-235 enrichment with depleted uranium and using boron oxide as a dissolving agent (DOE 2000).

On feeding the spent nuclear fuel into the melter, the uranium, plutonium, and other metals would be oxidized and dissolved in the molten glass. The oxidation of the metals would convert the lead oxide to metallic lead, which would sink to the bottom of the melter. Radioactive off-gases produced during this process would be filtered. The filters would be managed as high-level radioactive, low-level radioactive, or mixed waste, as appropriate.

Conversion of Lead to Lead Oxide: After decanting the glass, the melter would be recharged with boron oxide and, if necessary, lead oxide. Oxygen would be piped into the system to convert the metallic lead at the bottom of the melter back to lead oxide. Therefore, lead would be an oxygen carrier that would not leave the system.

Glass Waste Form: The resulting glass mixture would not have qualities necessary for long-term durability, so silicon oxide (glass frit) would need to be added to increase the durability of the high-level radioactive waste borosilicate glass. The silicon oxide would not be part of the initial melter charge because its properties are not conducive to rapid oxidation-dissolution of spent nuclear fuel. Unreduced lead oxide could limit the durability of the glass and increase volume, so carbon would be added to the melt to reduce the excess lead oxide (DOE 2000).

Technology Maturity: The GMODS process was developed by DOE for stabilization of radioactive waste. At this time, it has only been tested in small-scale laboratory experiments.

C.8 Chloride Volatility Process

Chloride volatility is an advanced treatment technology that was investigated at the Idaho National Engineering and Environmental Laboratory (NAS 1998). The process (1) uses the differences in the volatilities of chloride compounds to segregate major nonradiological constituents from spent nuclear fuel for the purpose of volume reduction, and (2) isolates the fissile material to produce a glass or ceramic waste form. A diagram of the chloride volatility process flow is shown in **Figure C–8**.

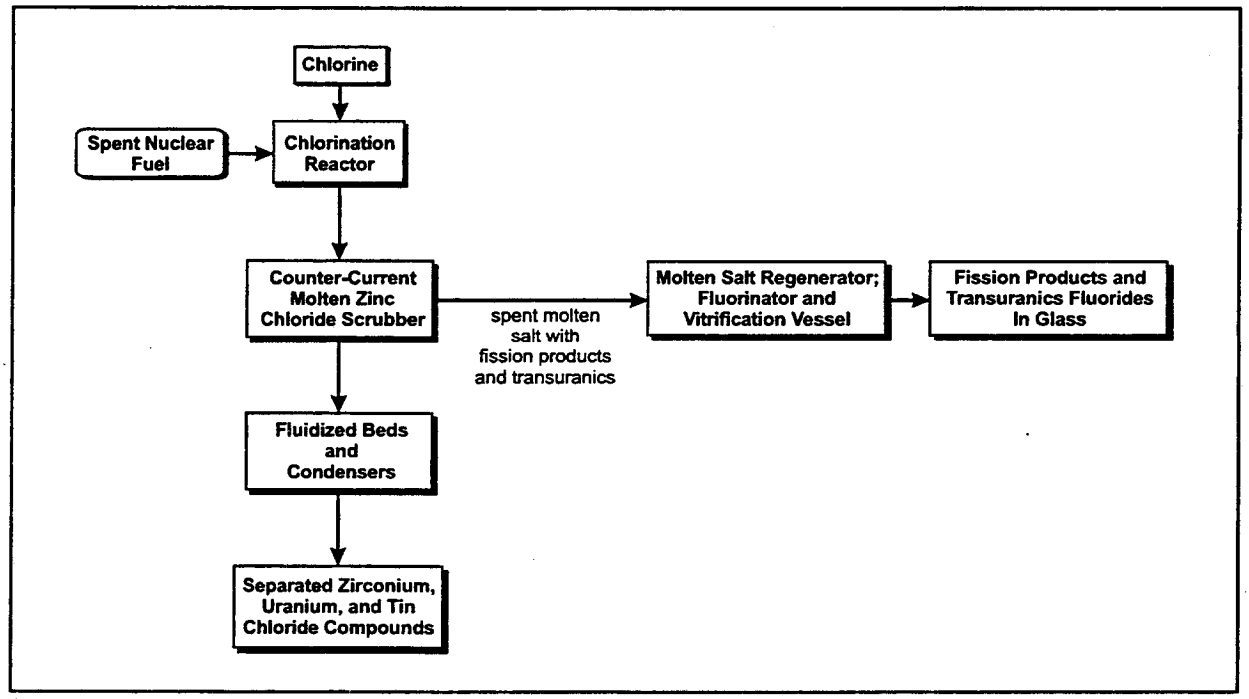

Figure C–8 Chloride Volatility Process Flow Diagram

The chloride volatility process would consist of four operations:

(1) A high-temperature chlorination step that would operate at approximately 1,500 °C (2,730 °F) and would convert fuel and cladding materials to gaseous chloride compounds

(2) A molten zinc chloride bed that would remove the transuranic chlorides and most of the fission products and would operate at approximately 400 °C (750 °F)

(3) A series of fluidized beds and condensers that would operate at successively lower temperatures to condense zirconium tetrachloride, uranium hexachloride, and stannous tetrachloride

(4) A zinc chloride regeneration/recycle process

The transuranic and fission product chlorides would be converted to either fluorides or oxides for final disposal. Argon carrier gas and unreacted chlorine gas would be recycled, the chlorine content adjusted, and the stream split and passed through the unit operations in a continuous closed loop. Periodic shutdowns of the coupled unit operations would occur for batch removal of fission product xenon and krypton gases from the carrier gas (such as by cryogenic distillation), batch transfer of the molten salt to the molten salt regenerator, and batch removal of nonradioactive constituents and uranium from the condensers.

The small quantity of fission-product/transuranic-product high-level radioactive waste would be converted into a waste form for repository disposal. The conversion steps to a glass or glass-ceramic form could involve fluorination and melting with glass frit additives, or conversion to oxides by heating at about 1,000 °C (1,830 °F) with boric acid.

In the chlorination step, the rate of reaction would be controlled by the feed rate of chlorine, and the temperature would be controlled by appropriate blending of argon gas with chlorine. An oxygen scavenger, such as carbon monoxide, would be added as needed to prevent formation of oxychlorides when oxides are present. A carbon dioxide absorption bed in the off-gas system would collect the carbon dioxide that would be formed. Zinc chloride would be used for the scrubber medium because its low melting point and favorable vapor pressure would permit its use to scrub the chlorinator off-gas at a low temperature, while its volatility at 725 °C (1,337 °F) would allow evaporative separation from the radioactive waste chlorides for subsequent recycle.

Theoretical chloride volatilities have been used to postulate the equipment sizing and operating parameters. Because of the lack of any experimental basis, significant concerns exist about the distribution of chloride compounds for multivalent elements such as uranium and plutonium. These concerns, in turn, lead to potential uncertainties in separation capabilities and overall flowsheet performance. The use of halides, either fluorides or chlorides, for the transuranic and fission product elements raises questions about the use of a glass or vitrified waste form. A proposal to use boric acid at about 1,000 °C (1,830 °F) allays some of those concerns (LITCO 1996).

Technology Maturity: The chloride volatility process has not progressed beyond the conceptual design stage. No laboratory experiments have been conducted.

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Appendix D

Sodium-Bonded Fuel Characteristics

APPENDIX D
SODIUM-BONDED FUEL CHARACTERISTICS

D.1 BACKGROUND

D.1.1 General Characteristics

The sodium-bonded spent nuclear fuel addressed in this environmental impact statement (EIS) is primarily from the operation of the Experimental Breeder Reactor-II (EBR-II) and Fermi-1 breeder reactors (a small percentage of the spent nuclear fuel is derived from other sources). Breeder reactors use two types of fuel: driver fuel, which is placed in the center of the reactor core, and blanket fuel, which is placed at the perimeter of the core. Driver fuel consists of highly enriched uranium alloy (alloy of uranium in zirconium or fissium[1]) fuel. (Natural uranium consists of mostly uranium-238, containing approximately 0.7 weight percent uranium-235; low-enriched uranium contains less than 20 weight percent uranium-235; highly enriched uranium contains greater than or equal to 20 weight percent uranium-235.) As a fissile material, uranium-235 is capable of undergoing fission (splitting into two major fragments called fission products) releasing energy and additional neutrons when struck by a neutron. This enriched uranium core produces the majority of the neutrons that power (drive) the reactor and breeding in the blanket, hence the name driver fuel. In the blanket region, uranium-238 from either natural uranium or depleted uranium, which has less than 0.3 weight percent uranium-235, capture neutrons to produce fissile materials, such as plutonium-239. In this manner, breeder reactors can produce (or breed) more fissile material than they consume.

The uranium in nuclear fuel is clad with a metal to protect it from chemical reactions with the coolant and to prevent the release of fission products to the coolant. Zirconium, stainless steel, and aluminum are common cladding materials. Most of the spent nuclear fuel analyzed in this EIS is clad with stainless steel.

Inside the cladding, the fuel is often in the form of a ceramic, an alloy that combines uranium with other metals such as zirconium, metallic uranium, or an oxide, carbide, nitride, or other form. The fuel can be fabricated as parallel plates, concentric tubes, bundles of rods or pins, or other designs. Each individual fuel item is referred to as a fuel element. Multiple fuel elements are typically combined into an assembly. Each assembly has mounting and lifting hardware, structures to direct coolant, and in some cases the capability to install neutron absorbing material and instrumentation. Most of the fuel elements addressed by this EIS are uranium alloy rods or pins. In order to improve the transfer of heat from the uranium matrix where the heat is generated to the cladding, the gap between the fuel and the cladding has been filled with a small amount of metallic sodium.

Usually a number of fuel assemblies make up a reactor core. Blanket assemblies placed around the reactor driver core for breeding or shielding are similar in design to driver fuel. An axial blanket may be placed above and below the reactor core and a radial blanket may be placed at the perimeter of the reactor core.

D.1.2 Recent Spent Nuclear Fuel Management Actions

In 1992, the Department of Energy (DOE) decided to phase out defense-related spent nuclear fuel reprocessing. Subsequently, the Department began to establish programs to manage DOE spent nuclear fuel that were no longer based on the production of strategic nuclear material. DOE identified the initial

[1] *Fissium is a mixture of noble metals (molybdenum, ruthenium, rhodium, palladium, zirconium, and niobium).*

components of this plan in the *Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement* (DOE 1995) (hereafter referred to as the Programmatic Spent Nuclear Fuel EIS). The Record of Decision for this EIS (60 FR 28680) stated, in part, that DOE would consolidate the management of its aluminum-clad spent nuclear fuel at the Savannah River Site (SRS), leave the Hanford production spent nuclear fuel at Hanford, and would consolidate nonaluminum-clad fuel at the Idaho National Engineering and Environmental Laboratory (INEEL). This Record of Decision was amended in March 1996 (61 FR 9441). The amended Record of Decision leaves all Fort St. Vrain spent nuclear fuel at the storage site in Colorado, all but sodium-bonded spent nuclear fuel at Hanford, and places restrictions on shipment schedules.

However, in the Programmatic Spent Nuclear Fuel EIS Record of Decision, DOE made no decisions on the technologies it would apply to the management of spent nuclear fuel at the designated storage sites. The Record of Decision stated that the selection of spent nuclear fuel stabilization technologies and the preparation of spent nuclear fuel for ultimate disposition would be the subject of site-specific and fuel-type-specific evaluations prepared in accordance with the National Environmental Policy Act (NEPA) and tiered from the Programmatic Spent Nuclear Fuel EIS (DOE 1995).

D.2 Inventory Overview

This EIS addresses a variety of spent nuclear fuel types that have one common characteristic, the presence of metallic sodium (or sodium and potassium). As a result of research, development, and demonstration activities associated with liquid metal fast breeder reactors, DOE has approximately 60 metric tons of heavy metal of spent nuclear fuel that contains metallic sodium. This EIS addresses a range of technologies that may be used to treat and manage this spent nuclear fuel for disposal. Based on composition, there are five broad categories of spent nuclear fuel to be considered: EBR-II driver spent nuclear fuel, EBR-II blanket, Fermi-1 blanket, Fast Flux Test Facility fuel, and miscellaneous spent nuclear fuel. While there are variations within each category, they may generally be described as follows:

EBR-II driver – This spent nuclear fuel is stainless steel clad highly enriched uranium in a uranium alloy, typically either fissium or zirconium. There are some variations in the specific cladding alloys, the enrichments, fuel compound alloy, dimensions, and burnup within this category. Also, there are small amounts of fuel experiments that use a different uranium compound, for example uranium carbide. This uranium carbide fuel type was added to the miscellaneous group.

EBR-II blanket – This spent nuclear fuel consists of stainless steel clad depleted uranium in a uranium metal form. There are various blanket designs: upper and lower axial, and inner and outer radial blankets. The primary difference between these blankets is dimension and burnup.

Fermi-1 blanket – This spent nuclear fuel consists of stainless steel clad depleted uranium in a uranium-molybdenum alloy. There are various blanket designs: upper and lower axial, and inner and outer radial blankets. The primary difference between these blankets is dimension, elements per assembly, and burnup. Fermi-1 blankets are similar to EBR-II blankets in enrichment, but differ in dimension (Fermi-1 elements are larger), burnup, and form (uranium metal versus uranium-molybdenum alloy).

Fast Flux Test Facility – This group of fuel includes both irradiated and fresh driver fuel. The fuel is either uranium zirconium or plutonium/uranium zirconium, with some containing plutonium/uranium carbide and nitride. This fuel is stainless steel-clad with various levels of enrichment.

Miscellaneous – This group includes experimental spent nuclear fuel from experiments irradiated in the Engineering Test Reactor and the Annular Core Research Reactor at Sandia National Laboratories/New Mexico, Oak Ridge National Laboratory fast reactor spent nuclear fuel, sodium research experiment spent nuclear fuel at SRS, and Westinghouse Atomic Power Division test reactor experiment at INEEL. There are small quantities of experimental fuel that have metallic sodium or potassium. This type of fuel is highly diverse and differs in cladding, uranium compound, enrichment, and burnup.

Table D–1 provides a summary of all DOE sodium-bonded spent nuclear fuel. It should be noted that the inventories reported in Table D–1 include 0.4 metric tons of heavy metal of EBR-II driver fuel and the 1.2 metric tons of EBR-II blanket fuel that are being treated as part of the electrometallurgical treatment demonstration program.

Table D–1 Overview of Sodium-Bonded Spent Nuclear Fuel Categories

Fuel Type	*Storage Volume (cubic meters)* [a]	*Total End of Life Fissile Mass (kilograms)*	*End of Life Mass Metric Tons of Heavy Metal*
EBR-II Driver	58 [b]	2,030	3.1
EBR-II Blanket	13	285	22.4
Fermi-1 Blanket	19	130	34.2
Fast Flux Test Facility	8 [b]	175	0.3
Miscellaneous	3 [b]	60	0.1
Total	**101**	**2,680**	**60**

[a] Volume refers to canister storage volume.
[b] A larger volume per unit mass for the driver fuel is required for the criticality control.
Source: ANL 1999.

By any measure, the majority of the spent nuclear fuel consists of EBR-II driver, EBR-II blanket, and Fermi-1 blanket fuel. **Table D–2** provides a summary of the fraction of spent nuclear fuel in each category by a variety of different measures. As shown, the percentages vary considerably depending upon the measure used for comparison.

Table D–2 Comparison of Sodium-Bonded Spent Nuclear Fuel by Different Measures

Fuel Type	*Storage Volume (percent)*	*Total End of Life Fissile Mass (percent)*	*End of Life Mass Metric Tons of Heavy Metal (percent)*
EBR-II Driver	58	75	5
EBR-II Blanket	13	11	37
Fermi-1 Blanket	19	5	57
Fast Flux Test Facility	8	7	0.5
Miscellaneous	3	2	less than 0.1
Total [a]	**100**	**100**	**100**

[a] Values may not add to exactly 100 percent due to rounding.

The radionuclide inventory of the spent nuclear fuel varies widely due to differences in the construction, function and operational history of the spent nuclear fuel. Therefore, radionuclide inventory estimates were developed for EBR-II driver fuel (including a separate estimate for the experimental driver fuel), EBR-II blanket, Fermi blanket, and Fast Flux Test Facility experimental fuel (SAIC 1999). **Table D–3** provides a summary of plutonium and sodium content for each fuel type.

Table D–3 Plutonium and Sodium Content in Sodium-Bonded Fuel

Spent Nuclear Fuel Type	*Plutonium Mass (kilograms)*	*Sodium Mass (kilograms)*
EBR-II Driver	19	83
EBR-II Blanket	250	176
Fermi-1 Blanket	7	365
Fast Flux Test Facility	3	7
Miscellaneous	0.10	31
Total	**279.10**	**662**

Table D–4 provides a list of principal radionuclide isotopes for each of the fuel types.

For each fuel type, principal radionuclide inventories were determined by considering all isotopes that, as a whole, contribute greater than 99.99 percent of the total dose in a case of accidental release. The dose estimates associated with each isotope intake were based on the effective committed dose equivalent factors provided in Federal Regulatory Guidance Report No. 11 (EPA 1988). Next, the list of isotopes was adjusted to include those isotopes with a boiling point less than 1,400° C (2,550° F), which is the maximum melt and dilute process temperature, and then isotopes of interest like hydrogen-3 (tritium), krypton-85, iodine-129, and uranium isotopes were added. The values in Table D–4 reflect the inventory of each isotope as of January 2000 (Liaw 1998).

The following sections provide a more detailed description of each category of spent nuclear fuel.

D.3 EBR-II Spent Nuclear Fuel

D.3.1 Reactor Background

EBR-II was a research and test reactor at Argonne National Laboratory-West (ANL-W) used to demonstrate the engineering feasibility of a sodium-cooled, liquid metal fast breeder reactor with a steam electric power plant and integral fuel cycle. It achieved initial criticality in September 1961 and continued to operate until September 1994. During its operation, numerous fuel designs were tested in EBR-II. The reactor operating power level was 62.5 megawatts-thermal.

D.3.2 Description of EBR-II Spent Nuclear Fuel

The EBR-II reactor consisted of an enriched driver core surround by depleted blanket assemblies. The reactor originally had an upper and lower axial blanket above and below the driver core, as well as a radial blanket around the perimeter of the driver core. It later operated with a radial blanket only. In addition, various experimental assemblies where placed into the core for testing. The following sections describe the driver fuel (including experiments) and blanket assemblies.

Table D–4 Principal Radionuclide Activities per Kilogram of Heavy Metal [a]

Elements	*Isotope*	*EBR-II Driver* [b]	*EBR-II Radial Blanket* [c]	*EBR-II Exp. Driver Fuel*	*Fermi-1 Blanket*	*FFTF Driver*
Tritium	H-3	1.23	0.00712	1.16	0.0000756	1.90
Carbon	C-14	0.000199	0.0000597	0.000954	1.05×10^{-8}	0.000674
Iron	Fe-55	4.87	0.0901	5.11	0.0000269	9.89
Cobalt	Co-60	0.481	0.0159	2.09	0.0000888	0.586
Nickel	Ni-63	0.229	0.00306	0.152	0.0000482	0.0491
Krypton	Kr-85	18.9	0.0520	16.5	0.000663	23.9
Strontium	Sr-90	197	0.807	171	0.0163	241
Yttrium	Y-90	197	0.807	171	0.0163	241
Ruthenium	Ru-106	1.51	0.135	2.67	7.02×10^{-10}	3.95
Rhodium	Rh-106	1.51	0.135	2.67	7.02×10^{-10}	3.95
Cadmium	Cd-113M	0.0464	0.000712	0.0511	2.86×10^{-6}	0.0659
Antimony	Sb-125	2.96	0.0231	2.98	2.92×10^{-6}	4.72
Tellurium	Te-125M	1.23	0.00951	1.23	1.20×10^{-6}	1.89
Iodine	I-129	0.0000735	1.44×10^{-6}	0.0000685	1.26×10^{-8}	0.0000898
Cesium	Cs-134	1.76	0.0134	1.93	6.66×10^{-9}	4.19
	Cs-137	221	1.73	199	0.0243	272
Barium	Ba-137M	209	1.64	188	0.0230	257
Cerium	Ce-144	2.96	0.0627	5.55	6.60×10^{-12}	9.88
Praseodymium	Pr-144	2.96	0.0627	5.55	6.60×10^{-12}	9.88
Promethium	Pm-147	82.6	0.407	80.2	0.0000810	128
Samarium	Sm-151	5.34	0.100	5.00	0.00131	6.49
Europium	Eu-154	0.567	0.00734	0.628	7.70×10^{-7}	0.969
	Eu-155	3.81	0.0481	3.97	0.0000671	5.28
Thorium	Th-228	0.0000514	1.55×10^{-7}	0.0000561	1.32×10^{-10}	0.0000739
Uranium	U-234	0.0404	1.33×10^{-6}	0.0371	3.20×10^{-8}	0.0407
	U-235	0.00131	3.77×10^{-6}	0.00120	7.48×10^{-6}	0.00123
	U-236	0.00121	4.24×10^{-6}	0.00104	1.09×10^{-7}	0.00141
	U-238	0.000111	0.000327	0.000120	0.000331	0.000117
Neptunium	Np-237	0.000289	8.37×10^{-6}	0.000287	2.28×10^{-7}	0.000401
Plutonium	Pu-238	0.166	0.00939	0.233	3.34×10^{-6}	0.304
	Pu-239	0.269	0.753	1.61	0.0134	0.739
	Pu-240	0.00911	0.0518	0.754	0.0000112	0.123
	Pu-241	0.00222	0.210	14.4	3.54×10^{-7}	1.60
Americium	Am-241	0.000391	0.0163	0.359	3.46×10^{-8}	0.0516
Americium	Am-242M	3.313×10^{-7}	0.000169	0.00218	7.84×10^{-14}	0.000140
Total	**Ci/kg** [d]	**957**	**7.18**	**884.1**	**0.0959**	**1,240**
Total heavy metal mass	metric tons	3.1	22.4	0.2 [e]	34.2	0.25

[a] Activities are in curies per kilogram of heavy metal, as of January 1, 2000.
[b] Inventory of Mark III driver fuel is bounding fuel for all EBR-II driver fuel type.
[c] Representative for all EBR-II blanket fuel.
[d] Curie per kilogram of heavy metal.
[e] EBR-II experimental driver fuel mass is a subset of EBR-II driver fuel.

D.3.2.1 Driver Fuel

Standard Driver Fuel

The driver fuel contains highly enriched uranium (enrichment of up to 78 weight percent). When the fuel is "spent," the enrichment (ratio of uranium-235 to total uranium) ranges between 55 percent and 76 percent.

Each driver fuel element has a metal rod (also called a fuel pin) about 36 centimeters (14 inches) long and less than 0.5 centimeters (0.2 inches) in diameter. A typical EBR-II driver fuel pin is a metal alloy of 90 percent uranium and 10 percent zirconium. This fuel pin and a small amount of metallic sodium were loaded into a 73.7-centimeter (29-inch) long stainless-steel tube (cladding) and welded shut, as shown in **Figure D–1**. This unit of fuel is called an "element." Sixty-one (in some 91) fuel elements were put together in a stainless-steel hexagonal "can" to make a fuel assembly approximately 2.3 meters (92 inches) long and 5.8 centimeters (2.3 inches) across. A typical fresh (unirradiated) driver fuel assembly contains 4.5 kilograms (9.9 pounds) of uranium and a typical irradiated fuel assembly contains 4.1 kilograms (9.0 pounds).

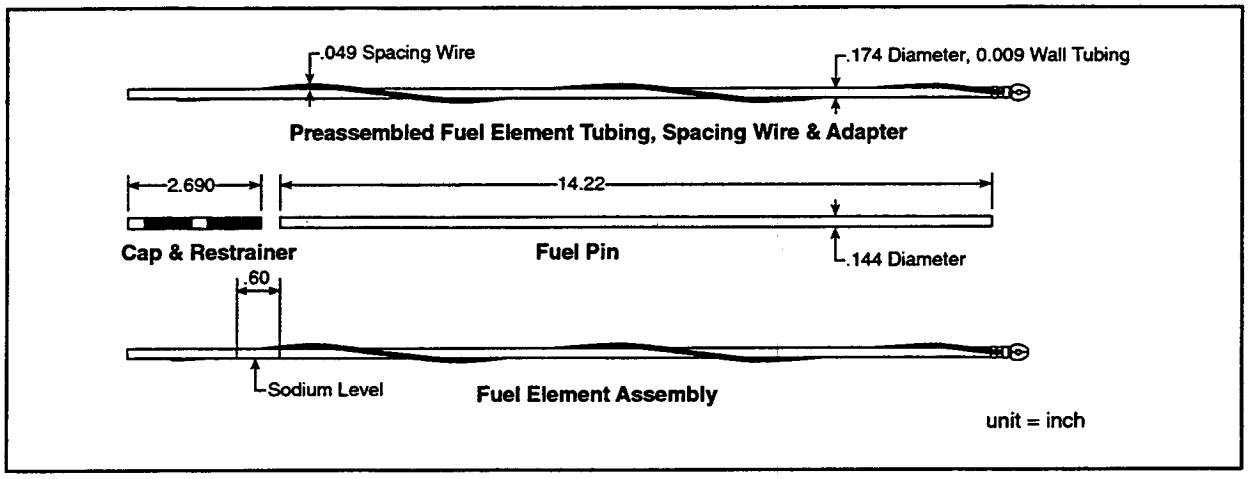

Figure D–1 Typical EBR-II Driver Element

The sodium inside driver and blanket elements improves the heat transfer from the fuel to the reactor coolant through stainless steel cladding. When the driver fuel is irradiated in the reactor for some period of time, the metallic pin swells until it reaches the cladding wall. Pores form throughout the fuel pin as it swells under pressure from the gaseous fission products. As these pores expand and connect to one another, the fission gases escape to a plenum in the fuel element just above the metallic fuel pin. As the gas escapes, the liquid sodium flows into these tiny pores, much like a sponge. As more pores form and grow, others are closed off from the fuel pin surface, including those containing sodium. Between 20 and 40 percent of the available sodium flows into the fuel pores and is inseparable from the uranium except by dissolving or melting the fuel. Further, during reactor operations, cesium-137 (an abundant radioactive fission product) dissolves in the sodium. Cesium, a reactive metal with chemical properties similar to sodium, remains with the sodium until the spent nuclear fuel is treated.

There have been numerous different fuel assemblies used in the EBR-II reactor, including a variety of experimental fuel. The types of standard spent nuclear fuel include Mark-I/IA, Mark-II/IIA, Mark-IIC/IICS, and Mark-III/IIIA. These different fuel types are quite similar, but differ in terms of dimensions, enrichment, fuel alloy, and cladding material. **Table D–5** shows the range of properties for EBR-II fuel, experimental fuel, and blanket elements.

Argonne National Laboratory has performed radionuclide projections individually for all of its spent nuclear fuel elements with the ORIGEN-RA depletion code and created a database containing inventory projections for all sodium-bonded spent nuclear fuel at ANL-W (Liaw 1998). The radionuclide inventory for a typical standard driver and experimental driver fuel element is presented in Table D–4. The driver fuel inventory is based on an average of the Mark-III elements, which are expected to have the highest inventory of the driver fuel. The EBR-II experimental driver inventory is based on the average of the experimental fuel elements that have not been processed. There may be individual elements with inventories that exceed this basis, but these inventories are well above the average for all driver assemblies.

Table D–5 Description of Unirradiated Typical EBR-II Driver and Blanket Fuel Elements

Property	*Standard Driver Fuel*	*Experimental Driver Fuel*	*Axial Blanket*	*Radial Blanket*
Element Description:				
Cladding material	SS-304L, SS-D-9, SS-316, SS-HT-9	SS-316, SS-HT-9, and SS-D-9	SS-304	SS-304
Clad outside diameter (inches)	0.179 – 0.23	0.17 - 0.29	0.38	0.49
Clad thickness (inches)	0.009 – 0.015	0.012 - 0.022	0.022	0.018
Element length (inches)	18 – 30	24 - 30	22	62
Fuel elements (or rods) per assembly	61 – 91	61	19	19
General Composition:				
Uranium alloy composition	U–5F [a] U–10Zr [b]	U–10Pu–10Zr [c] Pu/U-Carbide	Uranium metal	Uranium metal
Uranium-235 enrichment (percent)	67-78	Up to 93	0.2	0.2
Burnup (atom percent)	Up to 10	Up to 18	0.014	0.2
Sodium (g/element)	1.0 – 2.0	1.0 – 2.0	~ 3	~ 20

SS = Stainless steel.

[a] An alloy of 95 weight percent uranium and 5 weight percent fissium. Fissium consists of molybdenum, ruthenium, rhodium, palladium, zirconium, and niobium.

[b] An alloy of 90 weight percent uranium and 10 weight percent zirconium.

[c] An alloy of 80 weight percent uranium, 10 weight percent plutonium, and 10 weight percent zirconium.

Experimental Fuel

EBR-II has irradiated various types of different experimental driver fuel in support of its own and other liquid metal fast breeder reactor fuel development programs. Over 3,000 of these fuel elements still exist. Some of these experiments investigated the use of different fuel compositions including uranium-plutonium-zirconium alloy, plutonium-carbide, uranium-carbide and uranium-oxide. Table D–5 provides the range of data applicable for experiments. While the quantity of experimental spent nuclear fuel is relatively small, it is significant because of the associated potential unique requirements. Before this fuel can be treated, the carbide and oxide forms of the fuel may have to be reprocessed and converted to metallic forms.

D.3.2.2 Axial and Radial Blanket

The blanket assemblies were made from depleted uranium, a type of uranium in which most of the fissile uranium-235 has been removed, leaving 99.7 percent uranium-238. This type of uranium will fission, but not readily, and cannot be used alone to power a nuclear reactor. Early in EBR-II's history, the blanket assemblies surrounded or "blanketed" the reactor core to demonstrate the breeding of plutonium-239, another fissile material. However, in 1967 the breeding experiment was completed and the job of reconfiguring the reactor for its role as an irradiation test facility began. By 1972, the final blanket assemblies had been moved well

away from the core and replaced by a thick ring of stainless-steel reflector assemblies. In this configuration, the blanket assemblies provided shielding to protect structural materials from radiation emanating from the core.

Blanket assemblies are similar to the driver assemblies except that the individual blanket pins are larger. The blanket pins, made entirely from depleted uranium, are 1.1 centimeters (0.4 inches) in diameter, with three to five pins placed end-to-end to make a sodium-bonded blanket element 140 centimeters (55 inches) long. Since the blanket pins are a larger diameter and longer length, 19 blanket elements comprise a blanket assembly containing approximately 47 kilograms (103 pounds) of uranium. On average, about 99 percent of the uranium remains in the spent blanket assemblies with the remaining 1 percent having been converted to fission products and transuranic elements. The principal isotopes contributing to the activity of the axial and radial blanket assemblies are given in Table D–4.

Some of the EBR-II blanket assemblies have been in the reactor since it began operation more than 30 years ago. With the shutdown of EBR-II, these assemblies were unloaded from the reactor. In preparation for interim storage in the Radioactive Scrap and Waste Facility, they were cleaned to remove the few grams of sodium coolant that had adhered to the external surface as they were pulled out of the reactor.

D.3.2.3 Storage

Most of the fuel from the last seven years of EBR-II operation is presently stored in three different facilities at ANL-W: the Fuel Conditioning Facility, Hot Fuel Examination Facility, and Radioactive Scrap and Waste Facility. Previously, the spent nuclear fuel was shipped to the Idaho Nuclear Technology and Engineering Center (INTEC) (formerly Idaho Chemical Processing Plant) for reprocessing. However, INTEC ceased accepting the fuel in 1991 when a new uranium-zirconium alloy fuel, which could not be dissolved with INTEC's existing Chemical Processing Plant, went into full use at EBR-II. More than 6 metric tons (6.6 tons) of EBR-II fuel were processed at INTEC. When DOE stopped processing at INTEC in 1992, elements from some 500 EBR-II spent driver fuel assemblies of earlier design were left in storage pools located at INTEC. The spent nuclear fuel generated after shipments to INTEC ceased was stored at ANL-W in several facilities (Fuel Conditioning Facility, Hot Fuel Examination Facility, and Radioactive Scrap and Waste Facility).

D.4 FERMI-1 BLANKET

D.4.1 Reactor Background

The Enrico Fermi Atomic Power Plant[2] was designed and built at Monroe Beach, Michigan (30 miles southwest of Detroit) to demonstrate the feasibility of the fast breeder reactor for electric power production. Fermi-1 was a sodium cooled, fast reactor. Information was provided by Argonne National Laboratory, based upon EBR-I and EBR-II, to assist in the design of the Fermi-1 reactor. The reactor achieved initial criticality in 1963 and operated until September 1972. Fermi-1 was licensed for operation at a power level of 200 megawatts-thermal.

On October 5, 1966, Fermi-1 experienced a coolant blockage caused by a detached piece of zirconium liner. As a result, melting occurred in two subassemblies and the reactor was shutdown for three years and nine months. On July 18, 1970, the second Fermi-1 reactor core achieved criticality. New fuel and some of the original fuel was used for the second core. Termination of reactor operations in 1972 was not due to mechanical or technical problems, but rather due to lack of adequate financial support.

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[2]*The original name of the plant was the Enrico Fermi Atomic Power Plant. The numeral "1" was added to the name in 1969 after Detroit Edison Company began construction of Fermi-2. The plant is also known as Fermi, Fermi-1 or Enrico Fermi-1 (EF-1).*

D.4.2 Blanket Description

The reactor had two different blanket designs: axial blanket assemblies above and below the core, and radial blanket assemblies surrounding the core. The core assemblies (25.69 percent enriched fuel) were not bonded with sodium and are not part of the scope of this EIS. All blanket assemblies contain depleted uranium and contain a sodium bond between the uranium and the cladding. **Figure D–2** shows the radial blanket assembly. The inner and outer radial blanket assemblies had the same design and only differed in their placement in the reactor. The axial assemblies are similar, except they are shorter and have fewer, larger diameter pins. **Table D–6** provides data on both the axial and radial assemblies.

Table D–6 Description of Fermi-1 Blanket Elements and Assemblies

Property	*Axial Blanket*	*Radial Blanket*
Element Description:		
Cladding material	Stainless steel 304	Stainless steel 304
Clad outside diameter (inches)	0.443	0.443
Clad thickness (inches)	0.010	0.010
Uranium length (inches)	14	65
Fuel elements (pins or rods) per assembly	16 in upper blanket 16 in lower blanket	25
Assembly Description:		
Cross-section shape	Square	Square
Outside dimension (inches)	2.646	2.646
Wall thickness (inches)	0.096	0.096
Number of assemblies	403 [a]	559
General Composition:		
Uranium alloy composition	U–2.75 Mo [b]	U–2.75 Mo [b]
Uranium-235 enrichment (percent)	0.35	0.35
Sodium (grams/element)	5.5	20.7

[a] Includes both upper and lower axial blankets.
[b] An alloy of 97.25 percent depleted uranium and 2.75 percent molybdenum.

D.4.3 Storage

After the Fermi-1 reactor was permanently shutdown, the blanket assemblies were placed into 14 canisters and transported to INTEC in 1974 and 1975 in 14 shipments. The 14 canisters are made of stainless steel with a carbon steel basket inside. The canisters are 3.4 meters (11 feet, 2.5 inches) long and 65 centimeters (25.5 inches) in diameter. The canisters were filled with helium and seal welded. Twelve of the canisters contain the radial blanket assemblies and two of the canisters contain the shorter axial blanket assemblies.

D.5 Fast Flux Test Facility and Other Miscellaneous Fuel

As shown in Table D–2, the majority of the spent nuclear fuel addressed by this EIS is EBR-II driver, EBR-II blanket, or Fermi-1 blanket. However, there are small quantities of other spent nuclear fuel that also contain metallic sodium that are included in the scope of this EIS. These miscellaneous materials are described below.

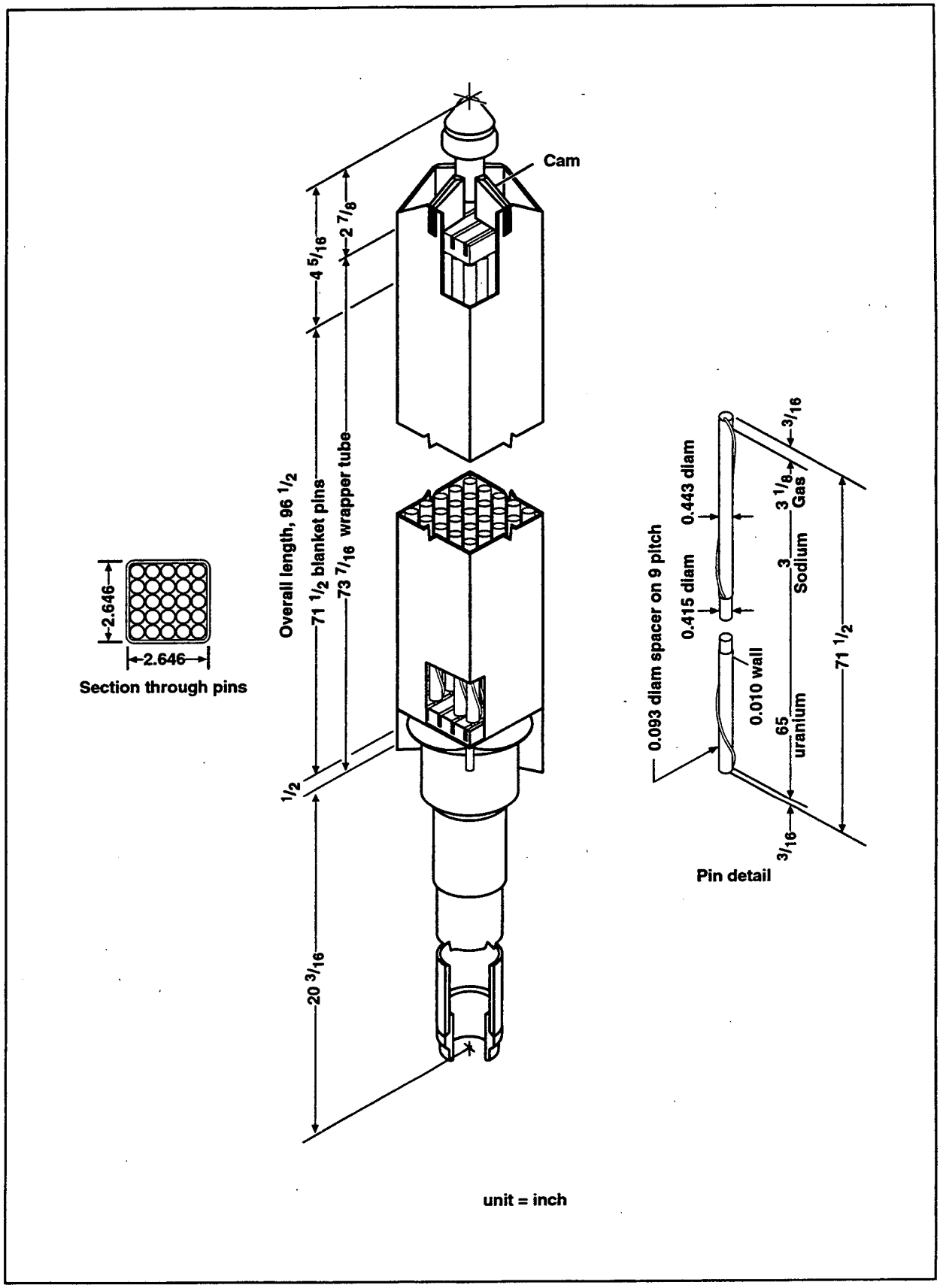


Figure D–2 Fermi-1 Radial Blanket Assembly

D.5.1 Fast Flux Test Facility

Background – The Fast Flux Test Facility, located on the Hanford Site near Richland, in southeastern Washington State, is a 400-megawatt thermal nuclear test reactor cooled by liquid sodium. It was built in 1978 and achieved initial criticality in 1980. The Fast Flux Test Facility was built to test plant equipment and fuel for the U.S. Government's liquid metal reactor development program. Although the facility is not a breeder reactor, this program demonstrated the technology of commercial breeder reactors. It was constructed to verify the safety and optimal performance of the key reactor systems and components. It was also intended to ensure the safety and best design of mixed oxide fuel, a mixture of uranium oxide and plutonium oxide.

The Fast Flux Test Facility successfully tested advanced nuclear fuel, materials, and safety designs. It also produced a large number of different medical isotopes, and made tritium for the U.S. fusion research program. Its operation also demonstrated the reactor's inherent safety features—most notably its ability during an emergency to remove reactor decay (residual) heat without pumps or any other mechanical system, simply based on its design. By contrast, current conventional water reactors require complex safety cooling and backup systems to remove their decay heat.

Description – Under normal operating conditions of the Fast Flux Test Facility, mixed oxide fuel with an enrichment of 20 to 30 percent plutonium was fabricated and inserted in the reactor core. However, the Fast Flux Test Facility also tested a number of experimental fuel types. The material included in the scope of this EIS is the sodium-bonded experimental fuel that was irradiated. **Table D–7** provides data on the sodium-bonded Fast Flux Test Facility spent nuclear fuel addressed by this EIS.

Storage – The Fast Flux Test Facility sodium-bonded spent nuclear fuel is currently in dry storage at the facility. The facility has no major vulnerabilities.

Inventory – There are just over 1,600 Fast Flux Text Facility rods (approximately 300 individual rods or elements and six assemblies consisting of 217 rods each) which are sodium-bonded totaling 0.32 metric tons of heavy metal. (Of this fuel, 0.07 metric tons of heavy metal, consisting of approximately 100 rods or elements and one assembly, are unirradiated fuel.) The radionuclide inventory of this spent nuclear fuel is presented in Table D–4.

D.5.2 Miscellaneous Fuel

Sandia National Laboratory Experiments

Background – A series of debris bed experiments were conducted at the Sandia National Laboratory's Annular Core Research Reactor from 1977 to 1985. These experiments were part of a program to study the "coolability" of debris beds that might be formed during reactor accidents. In the event of a severe accident in a sodium-cooled fast reactor, molten core materials may interact with liquid sodium and thus result in rapid quenching, freezing, and fragmentation. This fragmented debris may settle on horizontal surfaces within the reactor vessel to form debris beds. If the beds are subcritical, the debris will be heated by the radioactive decay of retained fission products. The possibility of damage to the pressure vessel and the containment, which prevent or mitigate the release of fission products as a consequence of the accident, depends on the extent to which natural cooling of the debris can be relied to remove decay heat from the bed. The debris bed experiments were the first "coolability" experiments to be conducted in-pile, using internally heated uranium dioxide and sodium.

Table D–7 Description of the Fast Flux Test Facility Sodium-Bonded Spent Nuclear Fuel

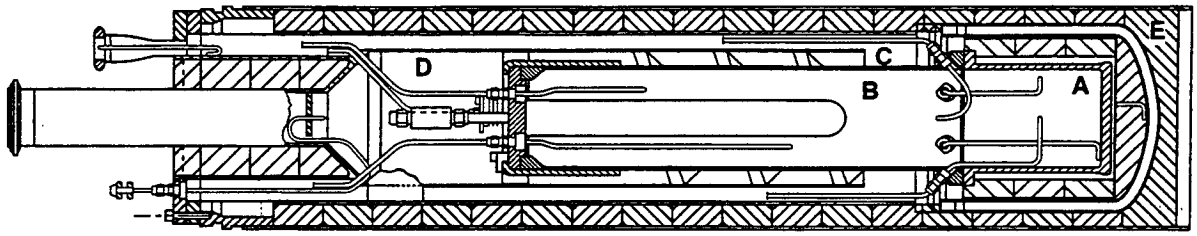
Property	*Fast Flux Test Facility Spent Nuclear Fuel*
Element Description:	
Shape	Round rod
Cladding material	Stainless steel 316 Stainless steel D9 Stainless steel HT9
Clad outside diameter (inches)	0.23 to 0.38
Clad thickness (inches)	0.022
Element length (inches)	93 to 120
Fuel pins or rods per assembly	217
Sodium (grams/element)	9 to 40
General Composition:	
Uranium alloy composition	Uranium-10 Zirconium [a] Uranium-10 Plutonium-10 Zirconium Plutonium/Uranium Carbide
Uranium-235 enrichment (percent)	0.2 to 24
Typical burnup (megawatt days/metric ton uranium)	68,000 to 140,000
Assembly Description:	
Rods per assembly	217
Assembly shape	Hexagon
Assembly width (inches)	4.567 flat to flat
Assembly height (inches)	144

[a] An alloy of 90 weight percent uranium and 10 weight percent zirconium.

Description – Each experiment consists of either a single or double containment within a helium chamber in the experiment section. Older experiments had a single containment, while newer ones were doubly contained. The uranium dioxide fuel, sodium, thermocouples, and in newer experiments, the insulated crucible are within the inner containment vessel. The uranium dioxide used in the experiments was produced by Los Alamos National Laboratory. The fuel was not irradiated prior to use in these experiments, nor was it melted during the experiments.

Figure D–3 provides a cut-away view of a typical debris bed experiment. As shown, these experiments are considerably different than the arrangement of sodium-bonded spent nuclear fuel. The fuel is just a small portion of the overall experiment structure. The fuel bed is held in a tantalum-tungsten alloy crucible with zirconia insulation. Each of the experiments is 10 centimeters (4 inches) in diameter and 50 centimeters (20 inches) long.

Storage – The seven debris bed experiments are stored dry at Sandia National Laboratories/New Mexico in Tech Area 5. The experiments are presently stored in seven "Dense Packs," a set of underground storage holes in Tech Area 5. There are no known vulnerabilities with this storage.



A bed in containment vessel
B sodium
C entrance for helium coolant to helical upflow heat exchanger
D exit plenum for hot helium
E typical solid insulation

Figure D–3 Typical Debris Bed Experiment

Inventory – The seven debris bed experiments have a total mass of 650 kilograms (1,433 pounds), of which only 34 kilograms (75 pounds) is highly-enriched uranium (93 percent uranium-235) and 20 kilograms (44 pounds) is metallic sodium. The sodium is interdispersed within the fuel debris. The burnup on this spent nuclear fuel is minor since the fuel had not been irradiated prior to these experiments.

The radionuclide inventory for these experiments was modeled as the EBR-II driver spent nuclear fuel on a heavy metal basis (see Table D–4). This is considered conservative because of the very low fuel burnup and the long cooling time (1977 to 1985, depending upon the experiment).

Westinghouse Atomic Power Division

Background – When the Engineering Test Reactor at INEEL was being taken to power, the activity of the primary reactor water rose abruptly. Within a few minutes after the rise began, the reactor received a slow setback which reduced power. Water chemistry analysis indicated a rupture in an experiment capsule. A small crack was found in one of the Westinghouse Atomic Power Division experiments (WAPD-49-AQ). There were 15 other similar experiment capsules in the reactor at the time. All of these capsules were removed from the reactor.

Description – The capsules have an overall length of 94.6 centimeters (37.25 inches) and are about 12.7 centimeters (5 inches) in diameter. Thirty centimeters (12 inches) of each capsule holds the fuel sample assembly. Each fuel sample assembly holds four fuel pins, each having a length of 14 centimeters (5.5 inches) and diameter of 0.9 centimeters (0.34 inches). The fuel pins contain uranium dioxide pellets (18 percent enriched). The oxide pellets have either one or two sheaths. The sheaths are made of either 304 stainless steel or zircaloy. The fuel pins that have two sheaths have a mixture of sodium and potassium between them. **Figure D–4** show the typical Westinghouse Atomic Power Division capsule arrangement.

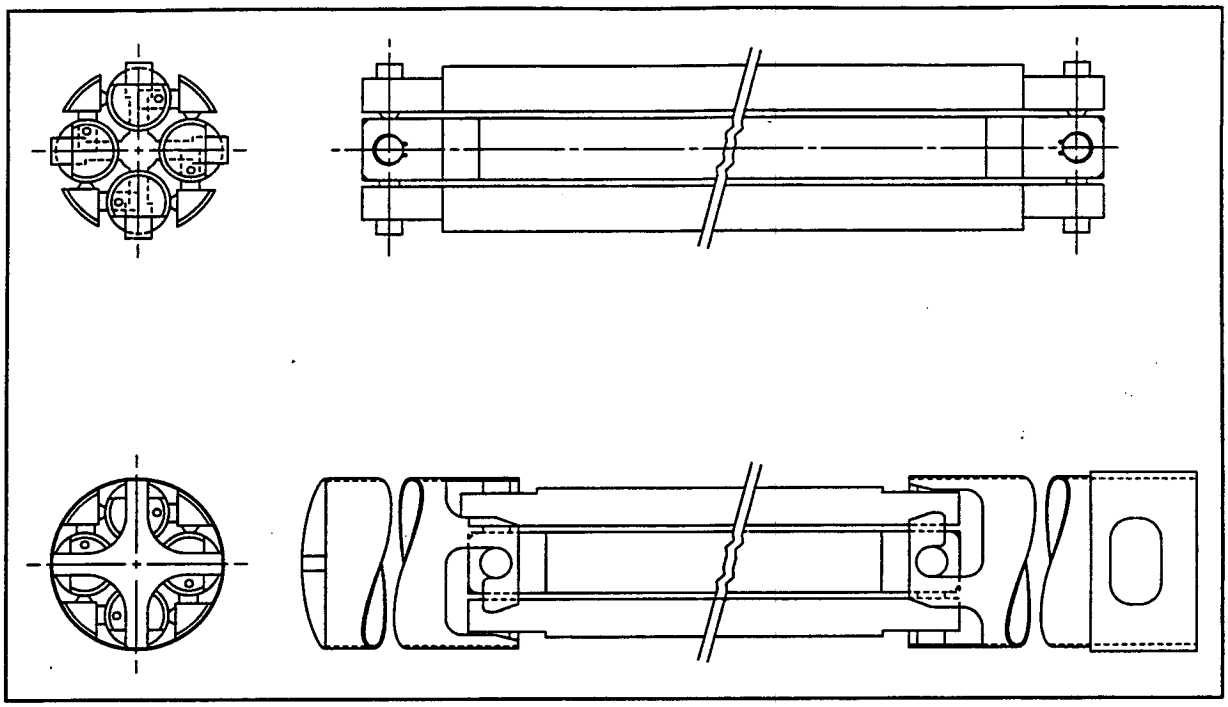


Figure D–4 Diagram of the Westinghouse Atomic Power Division Capsule

Storage – The Westinghouse Atomic Power Division spent nuclear fuel is currently stored in INTEC-603. There are a total of 22 experiments (i.e., pins). There are 4 experiments stored each in five aluminum cans and two capsules in the final can.

Inventory – The total inventory of the Westinghouse Atomic Power Division spent nuclear fuel is 6.6 kilograms (14.5 pounds) of uranium, at 18 percent enrichment. A radionuclide inventory of the Westinghouse Atomic Power Division spent nuclear fuel will be scaled conservatively from the EBR-II driver fuel inventory (see Table D–4) based upon heavy metal. This scaling approach is conservative because the experiments are fabricated with plutonium and uranium, have a lower enrichment, and have a lower burnup.

Oak Ridge National Laboratory Fast Reactor Spent Nuclear Fuel

Background – On August 12, 1998, the fuel elements were being sheared in half when a "sparkler-like reaction" was observed, lasting less than 30 seconds. This observed reaction was suspected of being an indication of sodium bonding on the spent nuclear fuel. This has not yet been confirmed. This spent nuclear fuel is included in this listing of sodium-bonded spent nuclear fuel in the event that it does prove to be sodium-bonded.

Description – The spent nuclear fuel is considered to be experimental EBR-II spent nuclear fuel elements. They are reported to be a uranium-carbide composition with stainless steel cladding. Figure D–1 shows the general configuration of EBR-II fuel, including experimental fuel. Table D–5 provides data on experimental EBR-II spent nuclear fuel.

Storage – This spent nuclear fuel is currently stored at the Oak Ridge National Laboratory in Building 3525, the Irradiated Fuel Examination Laboratory. The Irradiated Fuel Examination Laboratory is a two-story brick structure which contains hot cells. Disassembly and examination of fuel and components continue to be the mission of the facility. There are no identified vulnerabilities associated with this facility.

This spent nuclear fuel is stored in 4 containers in Building 3525. The containers are about 1.3 centimeters (0.5 inches) in diameter by 107 centimeters (42 inches) long.

Inventory – This spent nuclear fuel contains a total of 0.38 kilograms (0.84 pounds) of uranium, 0.35 kilograms (0.77 pounds) of which is uranium-235. Therefore, the enrichment is over 90 percent. This spent nuclear fuel also contains a total of 0.091 kilograms (0.20 pounds) of plutonium, 0.084 kilograms (0.18 pounds) of which is plutonium-239 or plutonium-241.

The radionuclide inventory for this small amount of material can be approximated by scaling the experimental spent nuclear fuel inventory (see Table D–4) based on heavy metal. This scaling approach is appropriate since this is an EBR-II experimental fuel.

Sodium Research Experiment at SRS

Background – The Sodium Research Experiment was a sodium-cooled, graphite-moderated reactor owned by the Atomic Energy Commission and Southern California Edison, Co. The Sodium Research Experiment achieved initial criticality in 1957 and was last operated in 1964. The Sodium Research Experiment operated at 20 megawatts-thermal until it was shut down in February 1964 for modification to permit an increase in power level to 30 megawatts-thermal. In December 1966, deactivation was announced.

Description – The Core I Sodium Research Experiment fuel was an unalloyed, uranium metal matrix, with a 2.8 percent uranium-235 enrichment stainless steel type 304 cladding, and sodium-potassium bonding. The Core I fuel contained seven rods per assembly. Core I was removed in 1959 after an incident resulted in the overheating and failure of one or more fuel in a number of fuel assemblies. The 26 undamaged fuel assemblies were shipped to Oak Ridge National Laboratory and were reprocessed. The assemblies that had damaged rods, along with miscellaneous fuel pieces retrieved from the reactor, were packaged into stainless steel canisters.

Core II assemblies were a thorium – 7.6 percent uranium alloy with a 92.3 percent uranium-235 enrichment, stainless steel type 304 cladding and sodium-potassium bonding. Core II fuel contained only five rods per assembly. Each rod contained 12 fuel slugs. Each fuel slug was 1.9 centimeters (0.75 inches) in diameter and 15.2 centimeters (6 inches) long. **Figure D-5** shows the typical assembly. The Core II fuel assemblies were removed from the reactor and placed into storage in 1964. This fuel was declad by Atomics International and shipped to SRS for reprocessing in 1976 and 1977.

In addition to the typical fuel, the Sodium Research Experiment also contained several types of experimental fuel. The experimental fuel addressed by this EIS is a uranium carbide fuel with a 9.8 percent uranium-235 enrichment, and stainless steel type 304 cladding.

Storage – The uranium carbide spent nuclear fuel addressed by this EIS is currently stored in the Receiving Basin for Offsite Fuel at the SRS. The Sodium Research Experiment spent nuclear fuel is stored in a can 8.9 centimeters (3.5 inches) in outer diameter and 366 centimeters (12 feet) long.

Inventory – This spent nuclear fuel contains a total of 43 kilograms (95 pounds) of uranium, 4.2 kilograms (9 pounds) of which is uranium-235. Therefore, the enrichment is 9.8 percent. This spent nuclear fuel also contains a total of 0.016 kilograms (0.035 pounds) of plutonium.

The radionuclide inventory for this small amount of material can be approximated by scaling the experimental spent nuclear fuel inventory (see Table D–4) based on heavy metal. This scaling approach is appropriate since this is a very small quantity of spent nuclear fuel with a burnup lower than the EBR-II spent nuclear fuel.

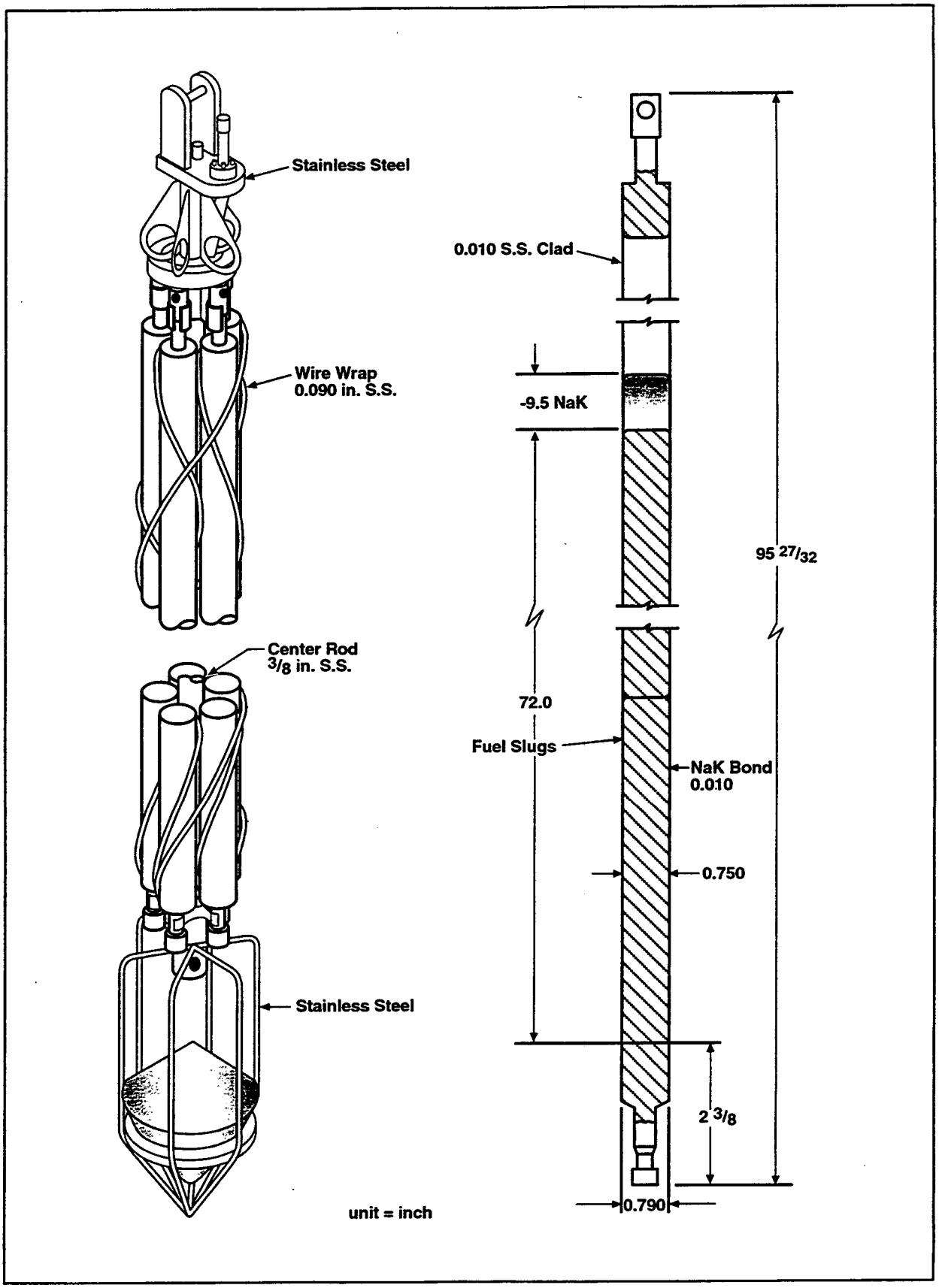


Figure D–5 Sodium Research Experiment Fuel Rod and Assembly Configuration

D.6 REFERENCES

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Appendix E

Evaluation of Human Health Effects from Normal Operations

APPENDIX E
EVALUATION OF HUMAN HEALTH EFFECTS FROM NORMAL OPERATIONS

E.1 INTRODUCTION

This appendix provides a brief general discussion on radiation and its associated health effects and describes the method and assumptions used for estimating the potential impacts and risks to individuals and the general public from exposure to the releases of radioactivity and hazardous chemicals during normal operations at the proposed facilities. Information regarding potential radiological impacts resulting from facility accidents is provided in Appendix F of this environmental impact statement (EIS).

This appendix presents numerical information using engineering and/or scientific notation. For example, the number 100,000 also can be expressed as 1×10^5. The fraction 0.00001 also can be expressed as 1×10^{-5}. The following chart defines the equivalent numerical notations that may be used in this appendix.

FRACTIONS AND MULTIPLES OF UNITS

Multiple	*Decimal Equivalent*	*Prefix*	*Symbol*
1×10^6	1,000,000	mega-	M
1×10^3	1,000	kilo-	k
1×10^2	100	hecto-	h
1×10	10	deka-	da
1×10^{-1}	0.1	deci-	d
1×10^{-2}	0.01	centi-	c
1×10^{-3}	0.001	milli-	m
1×10^{-6}	0.000001	micro-	μ
1×10^{-9}	0.000000001	nano-	n
1×10^{-12}	0.000000000001	pico-	p
1×10^{-15}	0.000000000000001	femto-	f
1×10^{-18}	0.000000000000000001	atto-	a

E.2 RADIOLOGICAL IMPACTS ON HUMAN HEALTH

Radiation exposure and its consequences are topics of interest to the general public. For this reason, this EIS places much emphasis on the consequences of exposure to radiation, provides the reader with background information on the nature of radiation, and explains the basic concepts used in the evaluation of radiation health effects.

E.2.1 Nature of Radiation and Its Effects on Humans

What Is Radiation?

Radiation is energy transferred in the form of particles or waves. Globally, human beings are exposed constantly to radiation from the solar system and the earth's rocks and soil. This radiation contributes to the natural background radiation that always surrounds us. Manmade sources of radiation also exist, including medical and dental x-rays, household smoke detectors, and materials released from nuclear and coal-fired power plants.

All matter in the universe is composed of atoms. Radiation comes from the activity of tiny particles within an atom. An atom consists of a positively charged nucleus (central part of an atom) with a number of negatively charged electron particles in various orbits around the nucleus. There are two types of particles in the nucleus: neutrons that are electrically neutral and protons that are positively charged. Atoms of different types are known as elements. There are more than 100 natural and manmade elements. An element has equal numbers of electrons and protons. When atoms of an element differ in their number of neutrons, they are called isotopes of that element. All elements have three or more isotopes, some or all of which could be unstable (i.e., decay with time).

Unstable isotopes undergo spontaneous change, known as radioactive disintegration or radioactive decay. The process of continuously undergoing spontaneous disintegration is called radioactivity. The radioactivity of a material decreases with time. The time it takes a material to lose half of its original radioactivity is its half-life. An isotope's half-life is a measure of its decay rate. For example, an isotope with a half-life of eight days will lose one-half of its radioactivity in that amount of time. In eight more days, one-half of the remaining radioactivity will be lost, and so on. Each radioactive element has a characteristic half-life. The half-lives of various radioactive elements may vary from millionths of a second to millions of years.

As unstable isotopes change into more stable forms, they emit electrically charged particles. These particles may be either an alpha particle (a helium nucleus) or a beta particle (an electron), with various levels of kinetic energy. Sometimes these particles are emitted in conjunction with gamma rays. The alpha and beta particles are frequently referred to as ionizing radiation. Ionizing radiation refers to the fact that the charged particle energy force can ionize, or electrically charge, an atom by stripping off one of its electrons. Gamma rays, even though they do not carry an electric charge as they pass through an element, can ionize its atoms by ejecting electrons. Thus, they cause ionization indirectly. Ionizing radiation can cause a change in the chemical composition of many things, including living tissue (organs), which can affect the way they function.

When a radioactive isotope of an element emits a particle, it changes to an entirely different element, one that may or may not be radioactive. Eventually a stable element is formed. This transformation, which may take several steps, is known as a decay chain. For example, radium, which is a member of the radioactive decay chain of uranium, has a half-life of 1,622 years. It emits an alpha particle and becomes radon, a radioactive gas with a half-life of only 3.8 days. Radon decays first to polonium, then through a series of further decay steps to bismuth, and ultimately to lead, which is a stable element. Meanwhile, the decay products will build up and eventually die away as time progresses.

The characteristics of various forms of ionizing radiation are briefly described below and in the box at right (see Chapter 6 for further definition):

Radiation Type	Typical Travel Distance in Air	Barrier
α	Couple of centimeters	Sheet of paper or skin's surface
β	Few meters	Thin sheet of aluminum foil or glass
γ	Very large *	Thick wall of concrete, lead, or steel
n	Very large	Water, paraffin, graphite

* *Would be infinite in a vacuum*

Alpha (α)

Alpha particles are the heaviest type of ionizing radiation. They can travel only a couple of centimeters in air. Alpha particles lose their energy almost as soon as they collide with anything. They can be stopped easily by a sheet of paper or by the skin's surface.

Beta (β)

Beta particles are much (7,330 times) lighter than alpha particles. They can travel a longer distance than alpha particles in the air. A high-energy beta particle can travel a few meters in the air. Beta particles can pass through a sheet of paper, but may be stopped by a thin sheet of aluminum foil or glass.

Gamma (γ)

Gamma rays (and x-rays), unlike alpha or beta particles, are waves of pure energy. Gamma rays travel at the speed of light. Gamma radiation is very penetrating and requires a thick wall of concrete, lead, or steel to stop it.

Neutrons (n)

Neutrons are particles that contribute to radiation exposure both directly and indirectly. The most prolific source of neutrons is a nuclear reactor. Indirect radiation exposure occurs when gamma rays and alpha particles are emitted following neutron capture in matter. A neutron has about one-quarter the weight of an alpha particle. It will travel in the air until it is absorbed in another element.

Units of Radiation Measure

During the early days of radiological experience, there was no precise unit of radiation measure. Therefore, a variety of units were used to measure radiation. These units were used to determine the amount, type, and intensity of radiation. Just as heat can be measured in terms of its intensity or effects using units of calories or degrees, amounts of radiation or its effects can be measured in units of curies, radiation absorbed dose (rad), or dose equivalent (roentgen equivalent man, or rem). The following summarizes those units (see also the definitions in the Glossary [Chapter 6]).

Curie

The curie, named after the French scientists Marie and Pierre Curie, describes the "intensity" of a sample of radioactive material. The rate of decay of 1 gram of radium is the basis of this unit of measure. It is equal to 3.7×10^{10} disintegrations (decays) per second.

Rad

The rad is the unit of measurement for the physical absorption of radiation. The total energy absorbed per unit quantity of tissue is referred to as absorbed dose (or simply dose). As sunlight heats pavement by giving up an amount of energy to it, radiation similarly gives up rads of energy to objects in its path. One rad is equal to the amount of radiation that leads to the deposition of 0.01 joule of energy per kilogram of absorbing material.

Radiation Units and Conversions to International System of Units

1 curie = 3.7×10^{10} becquerel
1 rad = 0.01 gray
1 rem = 0.01 sievert
1 gray = 1 joule per kilogram
1 becquerel = 1 disintegration per second

Rem

A rem is a measurement of the dose equivalent from radiation based on its biological effects. The rem is used in measuring the effects of radiation on the body as degrees centigrade are used in measuring the effects of sunlight heating pavement. Thus, 1 rem of one type of radiation is presumed to have the same biological effects as 1 rem of any other kind of radiation. This allows comparison of the biological effects of radionuclides that emit different types of radiation.

The units of radiation measure in the International System of Units are: becquerel (a measure of source intensity [activity]), gray (a measure of absorbed dose), and sievert (a measure of dose equivalent).

An individual may be exposed to ionizing radiation externally (from a radioactive source outside the body) or internally (from ingesting or inhaling radioactive material). The external dose is different from the internal dose because an external dose is delivered only during the actual time of exposure to the external radiation source, but an internal dose continues to be delivered as long as the radioactive source is in the body. The dose from internal exposure is calculated over 50 years following the initial exposure. Both radioactive decay and elimination of the radionuclide by ordinary metabolic processes decrease the dose rate with the passage of time.

Sources of Radiation

The average American receives a total of approximately 360 millirem per year from all sources of radiation, both natural and manmade, of which approximately 300 millirem per year are from natural sources. The sources of radiation can be divided into six different categories: (1) cosmic radiation, (2) terrestrial radiation, (3) internal radiation, (4) consumer products, (5) medical diagnosis and therapy, and (6) other sources (NCRP 1987). These categories are discussed in the following paragraphs.

Cosmic Radiation

Cosmic radiation is ionizing radiation resulting from energetic charged particles from space continuously hitting the earth's atmosphere. These particles and the secondary particles and photons they create comprise cosmic radiation. Because the atmosphere provides some shielding against cosmic radiation, the intensity of this radiation increases with the altitude above sea level. The average dose to people in the United States from this source is approximately 27 millirem per year.

External Terrestrial Radiation

External terrestrial radiation is the radiation emitted from the radioactive materials in the Earth's rocks and soils. The average dose from external terrestrial radiation is approximately 28 millirem per year.

Internal Radiation

Internal radiation results from the human body metabolizing natural radioactive material that has entered the body by inhalation or ingestion. Natural radionuclides in the body include isotopes of uranium, thorium, radium, radon, polonium, bismuth, potassium, rubidium, and carbon. The major contributor to the annual dose equivalent for internal radioactivity is the short-lived decay products of radon, which contribute approximately 200 millirem per year. The average dose from other internal radionuclides is approximately 39 millirem per year.

Consumer Products

Consumer products also contain sources of ionizing radiation. In some products, such as smoke detectors and airport x-ray machines, the radiation source is essential to the product's operation. In other products, such as televisions and tobacco, the radiation occurs as the product's function. The average dose from consumer products is approximately 10 millirem per year.

Medical Diagnosis and Therapy

Radiation is an important diagnostic medical tool and cancer treatment. Diagnostic x-rays result in an average exposure of 39 millirem per year. Nuclear medical procedures result in an average exposure of 14 millirem per year.

Other Sources

There are a few additional sources of radiation that contribute minor doses to individuals in the United States. The dose from nuclear fuel cycle facilities (e.g., uranium mines, mills, and fuel processing plants), nuclear power plants, and transportation routes has been estimated to be less than 1 millirem per year. Radioactive fallout from atmospheric atomic bomb tests, emissions of radioactive material from nuclear facilities, emissions from certain mineral extraction facilities, and transportation of radioactive materials contribute less than 1 millirem per year to the average dose to an individual. Air travel contributes approximately 1 millirem per year to the average dose.

Exposure Pathways

As stated earlier, an individual may be exposed to ionizing radiation both externally and internally. The different ways that could result in radiation exposure to an individual are called exposure pathways. Each type of exposure is discussed separately in the following paragraphs.

External Exposure

External exposure can result from several different pathways, all having in common the fact that the radiation causing the exposure is external to the body. These pathways include exposure to a cloud of radiation passing over the receptor (i.e., an individual member of the public), standing on ground that is contaminated with radioactivity, and swimming or boating in contaminated water. If the receptor departs from the source of radiation exposure, the dose rate will be reduced. It is assumed that external exposure occurs uniformly during the year. The appropriate dose measure is called the effective dose equivalent.

Internal Exposure

Internal exposure results from a radiation source entering the human body through either inhalation of contaminated air or ingestion of contaminated food or water. In contrast to external exposure, once a radiation

source enters the body, it remains there for a period of time that varies depending on decay and biological half-life. The absorbed dose to each organ of the body is calculated for a period of 50 years following the intake. The calculated absorbed dose is called the committed dose equivalent. Various organs have different susceptibilities to harm from radiation. The quantity that takes these different susceptibilities into account is called the committed effective dose equivalent, and it provides a broad indicator of the risk to the health of an individual from radiation. The committed effective dose equivalent is a weighted sum of the committed dose equivalent in each major organ or tissue. The concept of committed effective dose equivalent applies only to internal pathways.

Radiation Protection Guides

Various organizations have issued radiation protection guides. The responsibilities of the main radiation safety organizations, particularly those that affect policies in the United States, are summarized below.

International Commission on Radiological Protection

This Commission has the responsibility for providing guidance in matters of radiation safety. The operating policy of this organization is to prepare recommendations to deal with basic principles of radiation protection and to leave to the various national protection committees the responsibility of introducing the detailed technical regulations, recommendations, or codes of practice best suited to the needs of their countries.

National Council on Radiation Protection and Measurements

In the United States, this Council is the national organization that has the responsibility for adapting and providing detailed technical guidelines for implementing the International Commission on Radiological Protection recommendations. The Council consists of technical experts who are specialists in radiation protection and scientists who are experts in disciplines that form the basis for radiation protection.

National Research Council/National Academy of Sciences

The National Research Council is an organization within the National Academy of Sciences that associates the broad community of science and technology with the Academy's purposes of furthering knowledge and advising the Federal Government.

Environmental Protection Agency

The Environmental Protection Agency (EPA) has published a series of documents, *Radiation Protection Guidance to Federal Agencies*. This guidance is used as a regulatory benchmark by a number of Federal agencies, including the U.S. Department of Energy (DOE), in the realm of limiting public and occupational work force exposures to the greatest extent possible.

Limits of Radiation Exposure

Limits of exposure to members of the public and radiation workers are derived from International Commission on Radiological Protection recommendations. The EPA utilizes the National Commission on Radiological Protection and the International Commission on Radiological Protection recommendations and sets specific annual exposure limits (usually less than those specified by the Commission) in Radiation Protection Guidance to Federal Agency documents. Each regulatory organization then establishes its own set of radiation standards. DOE has established a set of limits for radiation workers in 10 CFR 835. **Table E–1** provides the various exposure limits set by DOE and the EPA for radiation workers and members of the public.

Table E–1 Exposure Limits for Members of the Public and Radiation Workers

Guidance Criteria (Organization)	*Public Exposure Limits at the Site Boundary*	*Worker Exposure Limits*
40 CFR 190 (EPA)	25 millirem per year (all pathways)	—
10 CFR 835 (DOE)	—	5.000 millirem per year [a]
DOE Order N441.1 (DOE)	—	2.000 millirem per year [a]
DOE Order 5400.5 (DOE) [b]	10 millirem per year (all air pathways) 4 millirem per year (drinking water pathway) 100 millirem per year (all pathways)	—
40 CFR 61 (EPA)	10 millirem per year (all air pathways)	—
40 CFR 141 (EPA)	4 millirem per year (drinking water pathways)	—

[a] Although these are limits (or levels) which are enforced by DOE, worker doses must still adhere to as low as reasonably achievable principles.

[b] Derived from 40 CFR 61, 40 CFR 141, and 10 CFR 20.

E.2.2 Health Effects

Radiation exposure and its consequences are topics of interest to the general public. To provide the background for discussions of impacts, this section explains the basic concepts used in the evaluation of radiation effects.

Radiation can cause a variety of damaging health effects in people. The most significant effects are induced cancer fatalities. These effects are referred to as "latent" cancer fatalities because the cancer may take many years to develop. In the discussions that follow, all fatal cancers are considered latent; therefore, the term "latent" is not used.

The National Research Council's Committee on the Biological Effects of Ionizing Radiation (BEIR) has prepared a series of reports to advise the U.S. Government on the health consequences of radiation exposures. *Health Effects of Exposure to Low Levels of Ionizing Radiation*, BEIR V (National Research Council 1990), provides the most current estimates for excess mortality from leukemia and other cancers that are expected to result from exposure to ionizing radiation. BEIR V provides estimates that are consistently higher than those in its predecessor, BEIR III. This increase is attributed to several factors, including the use of a linear dose response model for cancers other than leukemia, revised dosimetry for the Japanese atomic bomb survivors, and additional follow-up studies of the atomic bomb survivors and associated others. BEIR III employs constant, relative, and absolute risk models, with separate coefficients for each of several sex and age-at-exposure groups. BEIR V develops models in which the excess relative risk is expressed as a function of age at exposure, time after exposure, and sex for each of several cancer categories. The BEIR III models were based on the assumption that absolute risks are comparable between the atomic bomb survivors and the U.S. population. BEIR V models were based on the assumption that the relative risks are comparable. For a disease such as lung cancer, where baseline risks in the United States are much larger than those in Japan, the BEIR V approach leads to larger risk estimates than the BEIR III approach.

The models and risk coefficients in BEIR V were derived through analyses of relevant epidemiologic data that included the Japanese atomic bomb survivors, ankylosis spondylitis patients, Canadian and Massachusetts fluoroscopy (breast cancer) patients, New York postpartum mastitis (breast cancer) patients, Israeli tinea capitis (thyroid cancer) patients, and Rochester thymus (thyroid cancer) patients. Models for leukemia, respiratory cancer, digestive cancer, and other cancers used only the atomic bomb survivor data, although results of analyses of the ankylosis spondylitis patients were considered. Atomic bomb survivor analyses were based on revised dosimetry, with an assumed relative biological effectiveness of 20 for neutrons, and were restricted to doses less than 400 rads. Estimates of risks of fatal cancers, other than leukemia, were obtained by totaling the estimates for breast cancer, respiratory cancer, digestive cancer, and other cancers.

The National Council on Radiation Protection and Measurements (NCRP 1993), based on the radiation risk estimates provided in BEIR V and the International Commission on Radiological Protection Publication 60 recommendations (ICRP 1991), has estimated the total detriment resulting from low dose[1] or low dose rate exposure to ionizing radiation to be 0.00073 per rem for the general population and 0.00056 per rem for the working population. The total detriment includes fatal and nonfatal cancer and severe hereditary (genetic) effects. The major contribution to the total detriment is from fatal cancer and is estimated to be 0.0004 and 0.0005 per rem for the radiation workers and the general population, respectively. **Table E–2** provides the breakdown of the risk factors for both workers and the general population. Nonfatal cancers and genetic effects are less probable consequences of radiation exposure. To simplify the presentation of the impacts, estimated effects of radiation are calculated only in terms of latent cancer fatalities.

Table E–2 Nominal Health Effects Coefficients (Risk Factors) From Exposure to 1 Rem of Ionizing Radiation

Exposed Individual	*Fatal Cancer* [a,c]	*Nonfatal Cancer* [b]	*Genetic Disorders* [b]	*Total*
Worker	0.0004	0.00008	0.00008	0.00056
Public	0.0005	0.0001	0.00013	0.00073

[a] For fatal cancer, the health effect coefficient is the same as the probability coefficient. When applied to an individual, the units are the lifetime probability of a latent cancer fatality per rem of radiation dose. When applied to a population of individuals, the units are the excess number of cancers per person-rem of radiation dose.

[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. Genetic effects can be applied only to a population, not individuals.

[c] For high individual exposures (greater than or equal to 20 rem), the health factors are multiplied by a factor of 2.

Source: NCRP 1993.

The numerical estimates of fatal cancers presented in this EIS were obtained using a linear extrapolation from the nominal risk estimated for lifetime total cancer mortality, which is 0.1 gray (10 rad). Other methods of extrapolation to the low-dose region could yield higher or lower numerical estimates of fatal cancers. Studies of human populations exposed to low doses are inadequate to demonstrate the actual level of risk. There is scientific uncertainty about cancer risk in the low-dose region below the range of epidemiologic observation, and the possibility of no risk cannot be excluded (CIRRPC 1992).

Health Effect Risk Factors Used in This EIS

Health impacts from radiation exposure, whether from external or internal sources, generally are identified as "somatic" (i.e., affecting the exposed individual) or "genetic" (i.e., affecting descendants of the exposed individual). Radiation is more likely to produce somatic effects than genetic effects. The somatic risks of most importance are induced cancers. Except for leukemia, which can have an induction period (time between exposure to carcinogen and cancer diagnosis) of as little as 2 to 7 years, most cancers have an induction period of more than 20 years.

For a uniform irradiation of the body, the incidence of cancer varies among organs and tissues; the thyroid and skin demonstrate a greater sensitivity than other organs. Such cancers, however, also produce relatively low mortality rates because they are relatively amenable to medical treatment. Because fatal cancer is the most

[1] *Low dose is defined as the dose level where DNA repair can occur in a few hours after irradiation-induced damage. Currently, a dose level of about 0.2 grays (20 rad), or a dose rate of 0.1 milligrays (0.01 rad) per minute is considered to allow the DNA to repair itself in a short period (EPA 1994).*

probable serious effect of environmental and occupational radiation exposures, estimates of cancer fatalities rather than cancer incidence are presented in this EIS. The numbers of fatal cancers can be used to compare the risks among the various alternatives.

Based on the preceding discussion and the values presented in Table E–2, the fatal cancers to the general public during normal operations and for accidents in which individual doses are less than 20 rem are calculated using a health risk factor of 0.0005 per person-rem. For workers, a risk factor of 0.0004 excess fatal cancers per person-rem is used. (The risk factors are lifetime probabilities that an individual would develop a latent fatal cancer per rem of radiation.) This lower value reflects the absence of children (who are more radiosensitive than adults) in the work force. Nonfatal cancer and genetic disorders among the public are 20 and 26 percent, respectively, of the fatal cancer risk factor. For workers, the health risk estimators are both 20 percent of the fatal cancer risk factor. The nonfatal cancer risk factors are not used in this EIS.

The fatal cancer factors are used to calculate the statistical expectation of the effects of exposing a population to radiation. For example, if 100,000 people were each exposed to one time radiation dose of 100 millirem (0.1 rem), the collective dose would be 10,000 person-rem. The exposed population would then be expected to experience 5 additional latent cancer fatalities from the radiation (10,000 person-rem × 0.0005 lifetime probability of latent cancer fatalities per person-rem = 5 latent cancer fatalities).

Calculations of the number of excess fatal cancers associated with radiation exposure do not always yield whole numbers; calculations may yield numbers less than 1.0, especially in environmental impact applications. For example, if a population of 100,000 were exposed to a total dose of only 0.001 rem per person, the collective dose would be 100 person-rem, and the corresponding estimated number of latent cancer fatalities would be 0.05 (100,000 persons × 0.001 rem × 0.0005 latent cancer fatalities per person-rem = 0.05 latent cancer fatalities). The 0.05 means that there is one chance in 20 that the exposed population would experience one latent fatal cancer. In other words, the 0.05 latent cancer fatalities is the *expected* number of deaths that would result if the same exposure situation were applied to many different groups of 100,000 people. In most groups, no person (0 people) would incur a latent fatal cancer from the 0.001 rem dose each member would have received. In a small fraction of the groups, 1 latent cancer fatality would result; in exceptionally few groups, 2 or more latent cancer fatalities would occur. The *average* expected number of deaths over all the groups would be 0.05 latent cancer fatalities (just as the average of 0, 0, 0, and 1 is 1/4, or 0.25). The most likely outcome is 0 latent cancer fatalities.

The same concept is applied to estimate the effects of radiation exposure on an individual member of the public. Consider the effects of individual's exposure to a 360 millirem (0.36 rem) annual dose from all radiation sources. The probability that the individual will develop a latent fatal cancer from continuous exposure to this radiation over an average life of 72 years (presumed) is 0.013 (1 person × 0.36 rem per year × 72 years × 0.0005 latent cancer fatality risk per person rem = 0.013). This correlates to one chance in 77 that the individual would develop a fatal cancer.

E.3 Methodology for Estimating Radiological Impacts

The radiological impacts from releases during normal operation of the facilities used to treat and manage sodium-bonded spent nuclear fuel were calculated using Version 1.485 of the GENII computer code (PNL 1988). Site-specific input data were used including location, meteorology, population, and source terms. Section E.3.1 briefly describes GENII and outlines the approach used for normal operations.

E.3.1 GENII Computer Code

The GENII computer model, developed by Pacific Northwest National Laboratory, is an integrated system of various computer modules that analyze environmental contamination resulting from acute or chronic releases

to, or initial contamination in, air, water, or soil. The model calculates radiation doses to individuals and populations. The GENII computer model is well documented for assumptions, technical approach, method, and quality assurance issues. The GENII computer model has gone through extensive quality assurance and quality control steps, including comparing results from model computations with those from hand calculations and performing internal and external peer reviews (PNL 1988).

GENII code consists of several modules for various applications, see the code manual (PNL 1988) for details. For this EIS, only the ENVIN, ENV, and DOSE computer modules were used. The output of one module is stored in a file that can be used by the next module in the system. The functions of the three GENII computer modules used in this EIS are discussed below.

ENVIN

The ENVIN module of the GENII code controls the reading of input files and organizes the input for optimal use in the environmental transport and exposure module, ENV. The ENVIN code interprets the basic input, reads the basic GENII data libraries and other optional input files, and organizes the input into sequential segments based on radionuclide decay chains.

A standardized file that contains scenario, control, and inventory parameters is used as input to ENVIN. Radionuclide inventories can be entered as functions of releases to air or water, concentrations in basic environmental media (air, soil, or water), or concentrations in foods. If certain atmospheric dispersion options have been selected, this module would generate tables of atmospheric dispersion parameters that are used in later calculations. If the finite plume air submersion option is selected in addition to the atmospheric dispersion calculations, preliminary energy-dependent finite plume dose factors can be prepared as well. The ENVIN module prepares the data transfer files that are used as input by the ENV module; ENVIN generates the first portion of the calculation documentation—the run input parameters report.

ENV

The ENV module calculates the environmental transfer, uptake, and human exposure to radionuclides that result from the chosen scenario for the user-specified source term. The code reads the input files from ENVIN and then, for each radionuclide chain, sequentially performs the precalculations to establish the conditions at the start of the exposure scenario. Environmental concentrations of radionuclides are established at the beginning of the scenario by assuming decay of pre-existing sources, considering biotic transport of existing subsurface contamination, and defining soil contamination from continuing atmospheric or irrigation depositions. For each year of postulated exposure, the code then estimates the air, surface soil, deep soil, groundwater, and surface water concentrations of each radionuclide in the chain. Human exposures and intakes of each radionuclide are calculated for: (1) pathways of external exposure from finite atmospheric plumes; (2) inhalation; (3) external exposure from contaminated soil, sediments, and water; (4) external exposure from special geometries; and (5) internal exposures from consumption of terrestrial foods, aquatic foods, drinking water, animal products, and inadvertent intake of soil. The intermediate information on annual media concentrations and intake rates are written to data transfer files. Although these may be accessed directly, they are usually used as input to the DOSE module of GENII.

DOSE

The DOSE module reads the intake and exposure rates defined by the ENV module and converts the data to radiation dose.

E.3.2 Data and General Assumptions

To perform the dose assessments for this EIS, different types of data were collected and generated. This section discusses the various data, along with the assumptions made for performing the dose assessments in this EIS.

Dose assessments were performed for both members of the general public and workers around and at Argonne National Laboratory-West (ANL-W) and the Savannah River Site (SRS). These assessments were made to determine the incremental doses that would be associated with the alternatives addressed in this EIS. Incremental doses for members of the public were calculated (via GENII) for two different types of receptors:

- Maximally Exposed Offsite Individual—The maximally exposed offsite individual was assumed to be an individual member of the public located at a position on the site boundary that would yield the highest impacts during normal operations.

- Population—The general population living within 80 kilometers (50 miles) of the facility.

Meteorological Data

The meteorological data used for all normal operational scenarios discussed in this EIS were in the form of joint frequency data files. A joint frequency data file is a table listing the fractions of time the wind blows in a certain direction, at a certain speed, and within a certain stability class. The joint frequency data files were based on measurements taken over a period of several years at both the ANL-W and SRS sites.

Population Data

Population distributions were based on the 1990 Census of Population and Housing data (DOC 1992). Projections were determined for the year 2010 (representative year for operations) for areas within 80 kilometers (50 miles) of the release locations at ANL-W and SRS. The projected site-specific population in 2010, assumed to be representative of the population over the operational period evaluated, was used in the impact assessments. The population was spatially distributed on a circular grid with 16 directions and 10 radial distances up to 80 kilometers (50 miles). The grid was centered at the location from which the radionuclides were assumed to be released.

Source Term Data

The site- and process-specific source terms used to calculate the impacts of normal operations are provided in Section E.4.

Food Production and Consumption Data

Generic food consumption rates are established in the U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.109 (NRC 1977). This regulatory guide provides guidance for evaluating ingestion doses from consuming contaminated terrestrial and animal food products using a standard set of assumptions for crop and livestock growth and harvesting characteristics. In this EIS, food consumption rates were based on site-specific agricultural production rates and local diets.

Basic Assumptions

To estimate annual radiological impacts from normal operations, the following additional assumptions and factors were considered in using GENII:

- Radiological airborne gaseous and particulate emissions were assumed to be released to the atmosphere through the plant stacks. See Section E.4 for the specifics at each management facility.

- Ground contamination was based on dry deposition of radionuclides from normal operation releases, assuming no previously deposited radionuclides. Doses resulting from previously deposited radionuclides are accounted for in the baseline dose analysis, as presented in Chapter 3, and are not attributable to the processing of sodium-bonded spent nuclear fuel.

- Unless limited by the process duration, the inhalation exposure time to the plume was assumed to be per year for the maximally exposed offsite individual and the general population. Plume exposure parameters used in the GENII model for normal operations are provided in **Table E–3**.

- The exposed individual or population was assumed to have the characteristics and habits (e.g., inhalation and ingestion rates) of an adult human.

- A semi-infinite/finite plume model was used for air immersion doses. Other pathways evaluated were ground exposure, inhalation, and ingestion of food crops and animal products contaminated by deposition of radioactivity from the air.

- Resuspension of particulates was not considered because calculations of dust loading in the atmosphere show that this pathway is negligible compared to the other pathways.

- Reported release heights were used for atmospheric releases and were assumed to be the effective stack heights. The resultant doses were conservative, as use of the actual stack heights negates plume rise.

- The calculated doses were 50-year committed doses from 1 year of intake.

- Unless otherwise noted, radionuclide materials were considered to be released in the chemical form resulting in the largest radiological impact, thus maximizing the potential dose effect.

Table E–3 GENII Exposure Parameters to Plumes and Soil Contamination (Normal Operations)

Maximally Exposed Offsite Individual				*General Population*			
External Exposure		*Inhalation of Plume*		*External Exposure*		*Inhalation of Plume*	
Plume (hours)	*Soil Contamination (hours)*	*Exposure Time (hours)*	*Breathing Rate (cubic centimeters per second)*	*Plume (hours)*	*Soil Contamination (hours)*	*Exposure Time (hours)*	*Breathing Rate (cubic centimeters per second)*
6,136	6,136	8,766	270	4,383	4,383	8,766	270

Sources: PNL 1988, NRC 1977.

Worker doses associated with the processing alternatives were determined from historical data associated with similar operations. See Section E.4 for details.

E.3.3 Uncertainties

The sequence of analyses performed to generate the radiological impact estimates from normal operation include: (1) selection of normal operational modes, (2) estimation of source terms, (3) estimation of environmental transport and uptake of radionuclides, (4) calculation of radiation doses to exposed individuals, and (5) estimation of health effects. There are uncertainties associated with each of these steps. Uncertainties exist in the way the physical systems being analyzed are represented by the computational models and in the data required to exercise the models (due to measurement, sampling, or natural variability).

In principle, one can estimate the uncertainty associated with each source and predict the remaining uncertainty in the results of 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 results. However, conducting such a full-scale quantitative uncertainty analysis is neither practical nor a standard practice for a study of this type. Instead, the analysis is designed to ensure—through judicious selection of release scenarios, models, and parameters—that the results represent the potential risks. This is accomplished by making conservative assumptions in the calculations at each step. The models, parameters, and release scenarios used in the calculations are selected in such a way that most intermediate results and, consequently, the final estimates of impacts are greater than would be expected. As a result, even though the range of uncertainty in a quantity might be large, the value calculated for the quantity would be close to one of the extremes in the range of possible values, so the chance of the actual quantity being greater than the calculated value would be low (or the chance of the quantity being less than the calculated value if the criteria are such that the quantity has to be maximized). The goal of the radiological assessment for normal operation in this study is to produce results that are conservative.

The degree of conservatism in the calculated results is related closely to the range of possible values the quantity can have. This range is determined by what can be expected to occur realistically. Limitations on the processing of material (e.g., design capacity/processing rate, system availability, operational duration) provide upper limits to the quantity of spent nuclear fuel that can be processed in a given time, (e.g., annually). In many cases these restrictions were used to represent normal operating capacity, thus maximizing the amount of spent nuclear fuel that can be processed annually. Using these upper limits on processing rates provides a conservative estimate of the annual release of radionuclides during normal operation for each of the treatment techniques. These conservative release estimates were used to calculate the annual impacts presented for each alternative.

Details of some of the proposed treatment processes (e.g., melt and dilute) have not been finalized, yet the evaluation of worker doses can be performed using data associated with existing operations, where appropriate. While this introduces additional uncertainties in the estimation of worker exposures, many similarities between existing and proposed operations justify the use of this data. Among the features that justify use of existing data are the following: ANL-W and SRS are both committed to adhering to as low as reasonably achievable radiation protection practices; both sites have treated spent nuclear fuel under similar operating conditions; existing facilities (although modified in some cases) and existing protective features will be used; and any operational controls generated for new processes will be similar to existing operational (procedural) controls. These similarities between existing and proposed process controls mitigate some of the uncertainties inherent in estimating the impacts of processes yet to be finalized.

The radionuclide composition of source terms has been estimated conservatively. There are uncertainties in the radionuclide inventory which are proportional to the quantities of source terms that ultimately are released. For evaluation purposes, the inventory used is based on the spent nuclear fuel with the highest representative radionuclide content with no credit taken for further decay beyond that which occurred prior to the year 2000.

E.4 RADIOLOGICAL RELEASES TO THE ENVIRONMENT AND ASSOCIATED IMPACTS

This section summarizes the estimated radiological releases to the environment as well as resulting impacts associated with the various alternatives assessed in this EIS. Impacts to workers from these alternatives also are discussed. The methodology for estimating radiological impacts, associated input data, and analytical assumptions is provided in Section E.3.

E.4.1 Electrometallurgically Treat Blanket and Driver Fuel at ANL-W (Alternative 1)

Under this alternative, releases of radioactive material would occur during normal operational processing of the sodium-bonded fuel rods in the argon cell at the Fuel Conditioning Facility. Fuel assemblies would be disassembled in the Fuel Conditioning Facility air cell, and individual fuel elements then would be transferred to the argon cell for chopping and treatment in one of the electrorefiners. The entire inventory of gaseous fission products, mainly tritium and krypton-85, is assumed to be released during processing in the Fuel Conditioning Facility. The likelihood of release of radionuclides other than the gaseous fission products is very small. No radionuclides would be released from the packaged salt and packaged metallic waste material transferred from the Fuel Conditioning Facility to the Hot Fuel Examination Facility.

Estimated radioactive releases during normal operations at ANL-W were calculated using a conservative methodology. First, assumptions were made to estimate a maximum annual throughput of material to be processed at the Fuel Conditioning Facility. There would be two electrorefiners in the Fuel Conditioning Facility argon cell; blanket material would be treated in one of the two electrorefiners and driver material would be treated in the other. Both driver and blanket material could be processed each year. Based on an annual operational processing limit of 5,000 kilograms (11,023 pounds) of total heavy metal fuel material consisting of more than 600 kilograms (1,320 pounds) of heavy metal driver material, it was assumed that driver fuel would be processed at the maximum rate until all driver fuel was processed. In addition, it was assumed that the Experimental Breeder Reactor-II (EBR-II) fuel (driver and blanket) currently at ANL-W would be processed first. Using these assumptions, annual mass processing throughputs were developed for the purposes of estimating releases of radioactive material during normal operations, and are presented in **Table E–4**.

Radioactive releases from the Fuel Conditioning Facility argon cell during fuel treatment were estimated next. Radioactivity associated with the fuel to be processed was determined using the fuel radioactivity inventory values discussed in Appendix D. Estimated releases were based on a methodology developed in support of ANL-W's State of Idaho and National Emission Standards for Hazardous Air Pollutants air permitting activities, and agreed upon by the State of Idaho's Department of Environmental Quality (Bauer 1992). From this methodology, equilibrium concentrations in the argon cell (curies per cubic meter per curie processed) were calculated and applied to the inventory associated with the assumed annual throughputs shown in Table E–4. Annual radioactive releases to the atmosphere were calculated as the product of the radionuclide equilibrium concentrations in the argon cell, the annual argon cell atmosphere exhaust (74,400 cubic meters per year), and a conservative adjustment (0.00001) to account for the combined filtration of the two banks of high-efficiency particulate air filters that the cell exhaust must pass through before entering the environment. This filtration adjustment was not applied to tritium or krypton-85, as 100 percent of these radionuclides were assumed to be released.

The Fuel Conditioning Facility stack was modeled with an effective stack height of 60.96 meters (200 feet). This is the actual stack height, and for conservatism, no plume rise was included in the atmospheric dispersion modeling.

Table E–4 Annual Processing Assumptions for Estimation of Radiological Releases During Normal Operations Under Alternative 1 at ANL-W

Year of Processing	Driver Fuel (kilograms per year)		Blanket Fuel (kilograms per year)		Total Fuel (kilograms per year)		
	EBR-II [a]	Fast Flux Test Facility [b]	EBR-II [c]	Fermi-1	Driver	Blanket	Driver + Blanket
1	600	0	4,400	0	600	4,400	5,000
2	600	0	4,400	0	600	4,400	5,000
3	600	0	4,400	0	600	4,400	5,000
4	600	0	4,400	0	600	4,400	5,000
5	600	0	4,400	0	600	4,400	5,000
6	100	400	400	4,200	500	4,600	5,100
7	0	0	0	5,000	0	5,000	5,000
8	0	0	0	5,000	0	5,000	5,000
9	0	0	0	5,000	0	5,000	5,000
10	0	0	0	5,000	0	5,000	5,000
11	0	0	0	5,000	0	5,000	5,000
12	0	0	0	5,000	0	5,000	5,000
Totals (kilograms)	3,100	400	22,400	34,200	3,500	56,600	60,100

[a] EBR-II driver spent nuclear fuel consists of 1,100 kilograms of EBR-II driver spent nuclear fuel at ANL-W and 2,000 kilograms at INTEC.

[b] The Fast Flux Test Facility driver spent nuclear fuel consists of 250 kilograms of sodium-bonded Fast Flux Test Facility driver spent nuclear fuel at Hanford, 70 kilograms of unirradiated sodium-bonded Fast Flux Test Facility fuel, and 80 kilograms of miscellaneous spent nuclear fuel at INTEC, Sandia National Laboratory, SRS, and the Oak Ridge Reservation.

[c] EBR-II blanket spent nuclear fuel consists of EBR-II blanket spent nuclear fuel at ANL-W.

The dose resulting from the release of tritium (H-3) depends heavily on its chemical form. The inhalation dose from oxidized tritium is 25,000 times higher than for tritium in elemental form (ICRP 1982). The dose conversion factors used in the GENII code assume that tritium released to the environment is in the oxidized form and therefore are very conservative for releases that involve elemental tritium. Because of the argon atmosphere in the Fuel Conditioning Facility argon cell, releases of tritium to the cell atmosphere would not become oxidized, and stack releases of tritium most likely would be in the elemental form. The oxidation of elemental tritium to oxidized tritium has been shown to occur slowly in the environment, and for this EIS, the long-term dose from elemental tritium releases is conservatively estimated to be 1 percent of that for the oxidized form (DOE 1997). Therefore, the inventory of tritium for each year of electrometallurgical treatment processing at the Fuel Conditioning Facility was multiplied by a factor of 0.01 to convert them to an equivalent release of tritium oxide for use as input to the GENII code.

Radiological Gaseous Emissions

The estimated annual and total atmospheric releases are tabulated in **Table E–5**. This table lists only those radionuclides that resulted from a screening procedure to indicate potential significant dose contributions. The source term listed in Table E–5 for each of the first five years of processing (years 1 through 5) represents the source term that results in the highest annual offsite dose, and is therefore used for the maximum annual dose calculations. The project lifetime total values in Table E–5 represent the total estimated releases over the 12 years of processing at ANL-W.

Table E–5 Annual and Total Radiological Releases During Normal Operations Under Alternative 1 at ANL-W

Isotope [a]	Annual Releases (curies per year)			Project Lifetime Total (curies)
	Years 1 through 5	Year 6	Years 7 through 12	
H-3	770	680	0.38	4,530
C-14	1.7×10^{-12}	1.0×10^{-12}	2.3×10^{-16}	9.4×10^{-12}
Fe-55	1.4×10^{-8}	1.5×10^{-8}	5.8×10^{-13}	8.7×10^{-8}
Co-60	1.6×10^{-9}	9.7×10^{-10}	1.9×10^{-12}	8.8×10^{-9}
Ni-63	6.5×10^{-10}	1.7×10^{-10}	1.0×10^{-12}	3.4×10^{-9}
Kr-85	11,570	8,800	3.3	66,670
Sr-90	7.0×10^{-8}	5.2×10^{-8}	4.7×10^{-11}	4.0×10^{-7}
Y-90	7.0×10^{-8}	5.2×10^{-8}	4.7×10^{-11}	4.0×10^{-7}
Ru-106	3.2×10^{-8}	2.9×10^{-8}	7.6×10^{-17}	1.9×10^{-7}
Rh-106	3.2×10^{-8}	2.9×10^{-8}	7.6×10^{-17}	1.9×10^{-7}
Cd-113m	6.7×10^{-10}	5.2×10^{-10}	3.1×10^{-13}	3.9×10^{-9}
Sb-125	4.1×10^{-8}	3.6×10^{-8}	3.2×10^{-13}	2.4×10^{-7}
Te-125m	4.5×10^{-10}	3.9×10^{-10}	3.4×10^{-15}	2.6×10^{-9}
I-129	1.4×10^{-12}	9.7×10^{-13}	1.8×10^{-15}	8.2×10^{-12}
Cs-134	3.2×10^{-8}	4.0×10^{-8}	9.5×10^{-16}	2.0×10^{-7}
Cs-137	4.0×10^{-6}	2.9×10^{-6}	3.5×10^{-9}	0.000023
Ba-137m	3.8×10^{-6}	2.8×10^{-6}	3.3×10^{-9}	0.000022
Ce-144	1.2×10^{-9}	1.8×10^{-9}	1.9×10^{-20}	7.7×10^{-9}
Pr-144	1.2×10^{-9}	1.8×10^{-9}	1.9×10^{-20}	7.7×10^{-9}
Pm-147	2.9×10^{-8}	2.6×10^{-8}	2.3×10^{-13}	1.7×10^{-7}
Sm-151	2.1×10^{-9}	1.4×10^{-9}	3.7×10^{-12}	1.2×10^{-8}
Eu-154	2.1×10^{-10}	2.0×10^{-10}	2.2×10^{-15}	1.3×10^{-9}
Eu-155	1.4×10^{-9}	1.1×10^{-9}	1.9×10^{-13}	8.3×10^{-9}
Th-228	1.6×10^{-14}	1.3×10^{-14}	3.2×10^{-19}	9.1×10^{-14}
U-234	1.2×10^{-11}	7.8×10^{-12}	7.8×10^{-17}	6.7×10^{-11}
U-235	3.9×10^{-13}	2.6×10^{-13}	1.8×10^{-14}	2.3×10^{-12}
U-236	3.7×10^{-13}	2.6×10^{-13}	2.7×10^{-16}	2.1×10^{-12}
U-238	7.4×10^{-13}	7.7×10^{-13}	8.1×10^{-13}	9.4×10^{-12}
Np-237	3.9×10^{-13}	2.8×10^{-13}	2.1×10^{-15}	2.2×10^{-12}
Pu-238	2.9×10^{-10}	2.2×10^{-10}	3.4×10^{-14}	1.6×10^{-9}
Pu-239	7.1×10^{-9}	1.2×10^{-9}	1.4×10^{-10}	3.7×10^{-8}
Pu-240	4.7×10^{-10}	1.2×10^{-10}	1.1×10^{-13}	2.5×10^{-9}
Pu-241	1.9×10^{-9}	1.1×10^{-9}	3.6×10^{-15}	1.1×10^{-8}
Am-241	6.2×10^{-12}	1.8×10^{-12}	1.5×10^{-17}	3.3×10^{-11}
Am-242m	6.4×10^{-14}	9.3×10^{-15}	3.4×10^{-23}	3.3×10^{-13}
Totals	12,310	9,500	3.7	71,200

[a] The listed isotopes are present within the argon cell at the Fuel Conditioning Facility. Due to lack (scarcity) of oxygen in the argon cell, the tritium (H-3) released to the cell would be in molecular (elemental) form.

Population Impacts

The estimated annual radiological impacts due to the source term for the maximally exposed offsite individual and the general public residing within the 80 kilometer (50 mile) radius surrounding ANL-W are tabulated in **Table E–6**. Calculated impacts are shown for each year of processing as well as for each of the fuel types to be processed. Impacts are listed resulting from releases during processing EBR-II driver and blanket spent nuclear fuel during each of the first five years (years 1 through 5), processing some of all four fuel types during the sixth year (year 6), and processing Fermi-1 blanket spent nuclear fuel during each of the final six years (years 7 through 12). The impacts to the maximally exposed offsite individual and the surrounding population would result primarily from estimated releases of tritium (H-3) and krypton-85. Together, these two radionuclides would account for greater than 99.9 percent of the estimated impacts.

Table E–6 Annual Radiological Impacts to the Public From Operational Activities Under Alternative 1 at ANL-W

		Population		*Maximally Exposed Offsite Individual*	
Year(s) of Processing	*Spent Nuclear Fuel Type*	*Collective Dose (person-rem per year)*	*Latent Cancer Fatalities (number of cancers)*	*Dose (millirem per year)*	*Latent Cancer Fatality Risk*
1 - 5	EBR-II driver	0.0027	1.4×10^{-6}	0.00033	1.6×10^{-10}
	Fast Flux Test Facility driver	0	0	0	0
	EBR-II blanket	0.000083	4.2×10^{-8}	0.000010	5.0×10^{-12}
	Fermi-1 blanket	0	0	0	0
	All fuel, years 1 through 5	0.0028	1.4×10^{-6}	0.00034	1.7×10^{-10}
6	EBR-II driver	0.00046	2.3×10^{-7}	0.000054	2.7×10^{-11}
	Fast Flux Test Facility driver	0.0018	9.2×10^{-7}	0.00022	1.1×10^{-10}
	EBR-II blanket	7.6×10^{-5}	3.8×10^{-9}	9.1×10^{-7}	4.6×10^{-13}
	Fermi-1 blanket	9.1×10^{-7}	4.5×10^{-10}	1.1×10^{-7}	5.5×10^{-14}
	All fuel, year 6	0.0023	1.2×10^{-6}	0.00028	1.4×10^{-10}
7 - 12	EBR-II driver	0	0	0	0
	Fast Flux Test Facility driver	0	0	0	0
	EBR-II blanket	0	0	0	0
	Fermi-1 blanket	1.1×10^{-6}	5.4×10^{-10}	1.3×10^{-7}	6.5×10^{-14}
	All fuel, years 7 through 12	1.1×10^{-6}	5.4×10^{-10}	1.3×10^{-7}	6.5×10^{-14}

Total cumulative radiological impacts over the projected 13 years of operations under this alternative are tabulated in **Table E–7**. This table shows the sum of the calculated impacts to the maximally exposed offsite individual and the surrounding population over 12 years of fuel treatment.

Table E–7 Cumulative Maximum Radiological Impacts to the Public From Normal Operational Releases Under Alternative 1 at ANL-W

	Population		*Maximally Exposed Offsite Individual*	
	Collective Dose (person-rem)	*Latent Cancer Fatalities (number of cancers)*	*Dose (millirem)*	*Latent Cancer Fatality Risk*
Project total impacts [a]	0.0163	8.2×10^{-6}	0.00198	9.9×10^{-10}

[a] Total impacts are estimated for the 12-year duration of fuel treatment; there are no releases in the 13th year, i.e., only salt stabilization is performed.

Worker Impacts

Workers involved with electrometallurgical treatment activities at ANL-W could receive radiation doses during handling activities, such as receiving and unloading fuel casks, and transferring in-process waste material from the Fuel Conditioning Facility to the Hot Fuel Examination Facility. Doses received during in-cell activities likely would be very small. A maximally exposed worker dose estimate for this EIS is based on the regulatory limit of 5,000 millirem per year for radiation workers at DOE sites. If an individual worker received this dose each year of the 13 years of the electrometallurgical treatment project, the total worker dose would be 65,000 millirem with an associated risk of developing fatal cancer of 0.026.

However, actual worker doses are likely to be much lower than this maximum estimate. The ANL-W radiation control program incorporates the DOE Administrative Control Level of 2,000 millirem per year per person established for all DOE activities in DOE Order N441.1. In addition, ANL-W has established an administrative goal of 1,500 millirem per year to any individual. The general design goals at the Fuel Conditioning Facility, for example, were to maintain radiation fields below 0.5 millirem per hour at all workstations. This means that for an individual working at the Fuel Conditioning Facility for a full-time occupational work year of 2,000 hours, the annual dose would be 1,000 millirem.

Worker population doses were estimated by examining the type and duration of various operations performed by workers involved with the electrometallurgical treatment project. Doses can be estimated based on previous doses from similar activities at ANL-W. Based on information from ANL-W, the total worker population dose estimate is 22 person-rem per year, averaging out to an individual dose of 60 millirem per year for each of the 346 involved workers. If these estimates are extended out over the 13 years of operational activities (12 years of fuel treatment and a year of high-level radioactive waste conversion activities), the collective worker dose is 286 person-rem and the associated risk is 0.11 latent cancer fatalities. The estimated impacts to the worker population associated with this alternative are summarized in **Table E–8**.

Table E–8 Annual and Total Impacts to Workers From Operational Activities Under Alternative 1 at ANL-W

	Worker Population	
	Collective Dose (person-rem)	*Latent Cancer Fatalities*
Annual impacts	22	0.0088
Project total impacts [a]	319	0.13

[a] Total impacts are estimated for the 13-year processing duration, plus a year for deactivation activities at 33 person-rem.

E.4.2 Prepare Blanket Fuel and Electrometallurgically Treat Driver Fuel at ANL-W (Alternatives 2 Through 5)

In Alternatives 2 through 5, the blanket spent nuclear fuel assemblies would need to be prepared at the ANL-W facilities prior to packaging in high-integrity cans or processing in either the plutonium-uranium extraction (PUREX) process at SRS or the melt and dilute process at SRS or ANL-W. When the blanket spent nuclear fuel is to be processed at SRS, Alternative 3 (PUREX processing) and 5 (melt and dilute processing), the blanket spent nuclear fuel would be declad and cleaned at ANL-W in the argon cell of the Hot Fuel Examination Facility. Processing of the blanket spent nuclear fuel assemblies at ANL-W (Alternative 2, placing the blanket spent nuclear fuel in high-integrity cans, and Alternatives 4 and 6, melt and dilute) would not require decladding of the blanket spent nuclear fuel. This activity also would be performed in the argon cell of the Hot Fuel Examination Facility. The preparation of the blanket spent nuclear fuel under these alternatives would require only that the fuel be cut into segments and cleaned (see Appendix C for details). The following discussion addresses the radiological impact of normal operations at ANL-W for the preparation of the blanket spent nuclear fuel elements and the electrometallurgical treatment of the driver spent nuclear fuel elements. This analysis is applicable to Alternatives 2 through 5.

Gaseous Emissions

Blanket spent nuclear fuel preparation would occur at the Hot Fuel Examination Facility. These activities would cause gaseous fission products to be released into the argon cell. As stated in Section E.4.1, krypton-85 and elemental tritium are the most prevalent gaseous radionuclides that would be released to the environment. The released tritium (H-3) into the cell would not be oxidized because of a very low presence of oxygen and humidity in the argon cell. The argon cell also contains an equilibrium concentration of other radionuclide isotopes. Appendix E, Section E.4.1, provides a list of various isotopes that are present in the argon cell in nanocuries (10^{-9} curies) and are released to the atmosphere through the facility stack, along with the krypton-85 and elemental tritium. The maximum released curies of radioactive gaseous emissions occurs when preparation of the blanket spent nuclear fuel and chopping of the driver spent nuclear fuel (for electrometallurgical treatment processing) are performed simultaneously. This simultaneous operation was estimated to occur over a six-year period starting in 2003. Based on a blanket spent nuclear fuel preparation throughput of 10 metric tons of heavy metal and an electrometallurgical treatment process rate of about 0.6 metric tons of heavy metal of driver spent nuclear fuel elements annually, at most about 809 curies of elemental tritium and 11,860 curies of krypton-85 would be released to the atmosphere annually; see **Table E–9**. This release rate would last about two years, or until all of the EBR-II blanket spent nuclear fuel is processed; then the release rate would drop during the processing of the Fermi-1 blanket spent nuclear fuel (the release rate for the processing of 10 metric tons of heavy metal of Fermi-1 blanket spent nuclear fuel is less than 1 curie of elemental tritium and 6.6 curies of krypton-85). The gaseous fission products generated during treatment processes at the Hot Fuel Examination Facility would be released to the atmosphere through the 31-meter (94-foot) facility stack.

Table E–9 Maximum Annual Radiological Gaseous Emission From Activities Associated With Alternatives 2 Through 5 at ANL-W

Spent Nuclear Fuel Type	*Facility*	*Maximum Processing Rate (metric tons of heavy metal per year)*	*Duration (years)*	*Annual Release (curies)*	
				Tritium [a]	*Krypton-85*
Driver fuel	Fuel Conditioning Facility	0.6	6	738	11,340
EBR-II blanket fuel	Hot Fuel Examination Facility	10	2.4	71.2	520
Fermi-1 blanket fuel	Hot Fuel Examination Facility	10	3.6	0.76	6.6
Maximum annual release [b]			2.4	809	11,860

[a] Elemental tritium; about 1 percent of tritium was assumed to be in oxidized form. See discussion in Section E.4.1.
[b] Maximum annual release occurs during concurrent processing of EBR-II driver and blanket spent nuclear fuel at ANL-W.

Population Impacts

The doses to the maximally exposed offsite individual and the general public residing within the 80 kilometer (50 mile) radius surrounding ANL-W are presented in **Table E–10**. As stated in Section E.4.1 the dose resulting from the release of tritium is highly dependent upon its chemical form. The doses presented in Table E–10 result from releases that are assumed to be 1 percent oxidized tritium, the same assumption used in the analysis of Alternative 1. These impacts are calculated for the preparation of the blanket spent nuclear fuel assemblies, for the processing of the driver spent nuclear fuel assemblies using the electrometallurgical treatment process, and the maximum total impacts. The maximum annual impact is associated with the concurrent treatment of EBR-II driver and blanket spent nuclear fuel. Under alternatives 2 through 5, treatment of the driver and blanket spent nuclear fuel does not begin at the same time. Electrometallurgical treatment of the driver spent nuclear fuel is expected to begin three years before the beginning of treatment of the blanket spent nuclear fuel (see the discussion for each alternative in Chapter 4). This results in the nine-year treatment duration identified in Table E–10, where only driver spent nuclear fuel is treated in the first three years; both driver and blanket spent nuclear fuel are treated in years four through six; and only blanket spent nuclear fuel is treated in the final three years. In Alternative 4, which includes melt and dilute processing of blanket spent nuclear fuel at ANL-W, the gaseous radionuclides, which result in over 99 percent of the offsite dose (tritium, krypton, and iodine), are released during the declad and clean process. It is the six-year duration of this portion of the melt and dilute process that was used as the time frame for modeling the operation releases from the treatment of blanket spent nuclear fuel for Alternative 4. As stated earlier, treatment of Fermi-1 spent nuclear fuel at ANL-W would have a negligible contributing impact. These impacts are applicable to the processing of blanket and driver spent nuclear fuel under Alternatives 2, 3, 4, and 5 at ANL-W.

Table E–10 Annual and Total Radiological Impacts to the Public From Normal Operational Releases Under Alternatives 2 Through 5 at ANL-W

		Population		*Maximally Exposed Offsite Individual*	
Year(s) of Processing	*Spent Nuclear Fuel Type*	*Collective Dose (person-rem per year)*	*Latent Cancer Fatalities (number of cancers)*	*Dose (millirem per year)*	*Latent Cancer Fatality Risk*
1-3	Driver	0.0027	1.4×10^{-6}	0.00033	1.6×10^{-10}
	Blanket	0	0	0	0
	All fuel	0.0027	1.4×10^{-6}	0.00033	1.6×10^{-10}
4-5	Driver	0.0027	1.4×10^{-6}	0.00033	1.6×10^{-10}
	EBR-II blanket	0.00028	1.4×10^{-7}	0.000048	2.4×10^{-11}
	All fuel	0.0030	1.5×10^{-6}	0.00038	1.9×10^{-10}
6	Driver	0.0023	1.2×10^{-6}	0.00028	1.4×10^{-10}
	EBR-II blanket	0.00011	5.6×10^{-8}	0.000019	9.6×10^{-12}
	Fermi-1 blanket	1.9×10^{-6}	9.7×10^{-10}	3.3×10^{-7}	1.6×10^{-13}
	All fuel	0.0024	1.2×10^{-6}	0.00030	1.7×10^{-10}
7-9	Driver	0	0	0	0
	Fermi-1 blanket	3.2×10^{-6}	1.6×10^{-9}	5.5×10^{-7}	2.7×10^{-13}
	All fuel	3.2×10^{-6}	1.6×10^{-9}	5.5×10^{-7}	2.7×10^{-13}
Project total [a]		0.0165	8.3×10^{-6}	0.0021	1.0×10^{-9}

[a] Maximum annual radiological impacts occur during two years of concurrent EBR-II driver and blanket spent nuclear fuel processing.

Worker Impacts

The worker activities under Alternatives 2 through 5 at ANL-W would be similar to those under Alternative 1. Therefore, the annual worker dose and the worker population dose would be similar to those provided in Section E.4.1. The project total dose is provided in Section 4.4.4.1.

E.4.3 PUREX Processing at SRS (Alternative 3)

PUREX processing at F-Canyon would release radioactive gaseous fission products during treatment of about 57 metric tons of heavy metal of EBR-II and Fermi-1 blanket spent nuclear fuel. Since declad and cleaned blanket spent nuclear fuel would be packaged and sent to SRS, no additional gaseous fission products would be expected to be present in that fuel. However, it was assumed conservatively that the gaseous fission products in the blanket spent nuclear fuel would remain within the fuel matrix and would be released to the environment (from the facility stack, 60 meters [198 feet] high) during PUREX processing at SRS. As a result, there would be incurred doses to the public associated with PUREX operations. The duration of PUREX operations was estimated to be six months, based on the F-Canyon's throughput and consistent with assumptions made for the treatment duration of a similar-type fuel at SRS in the *SRS Spent Nuclear Fuel Management Final Environmental Impact Statement* (DOE 2000).

Gaseous Emissions

According to SRS Spent Nuclear Fuel EIS data (DOE 1997), tritium (H-3) and krypton-85 are the only isotopes that would be expected to be released during PUREX processing operations. Based on the assumption that the entire fission gas inventory would remain within the fuel matrix after the decladding and cleaning process, it was assumed that the inventory of krypton-85 and tritium would be released. Using the gaseous fission product inventory provided in Appendix D for the EBR-II and Fermi-1 blanket spent nuclear fuel, the potential airborne radiological release quantities were estimated and are presented in **Table E–11**. This inventory was used to calculate the population doses from air emissions.

Table E–11 Estimated Incremental Releases of Radiological Air Emissions and Liquid Effluent During Normal Operations of PUREX Processing Under Alternative 3 at SRS

Isotope	*Releases to Air (curies)*	*Releases to Liquid (curies)* [a]
H-3	162	1.54
Kr-85	1,188	-
Sr-89/90	-	0.000031
Cs-137	-	0.0022
U-234	-	0.000085
U-235	-	0.000011
U-238	-	0.00019
Pu-238	-	0.000016
Pu-239	-	7.76×10^{-6}

[a] Estimated curies using the information provided in the SRS Environmental Data for 1997 (Arnett and Mamatey 1998).

Liquid Effluent

PUREX processing is the only process among the alternatives considered that would release measurable radioactive nuclides to the surface water. This release would occur through the cooling water system. The expected radiological effluent from processing declad and cleaned blanket spent nuclear fuel at F-Canyon were estimated based on the measured data from various effluent streams at F-Area, as presented in the SRS

Environmental Report and Data for 1997 (Arnett and Mamatey 1998). Since the mechanism associated with releases of liquid effluent from PUREX processing at F-Canyon is essentially the same for almost every fuel type processed, the F-Area 1997 effluent data were used to conservatively represent the potential releases from a six-month operation of F-Canyon. Table E–11 lists the radionuclides and their corresponding curies that are estimated to be released during PUREX processing of blanket spent nuclear fuel.

Population Impacts

Estimated annual radiological impacts associated with the F-Canyon PUREX operations for the maximally exposed offsite individual and the general population residing within the 80-kilometer (50-mile) radius surrounding F-Canyon are presented in **Table E–12**. This table provides the radiological doses to the public from air emissions and liquid effluent separately. According to the SRS Environmental Report, a maximally exposed offsite individual associated with liquid releases is an individual who lives downriver of SRS 365 days per year, drinks 2 liters of untreated water per day from the Savannah River, consumes a large amount of Savannah River fish, and spends the majority of time on or near the river. The general population liquid effluent dose is calculated for the discrete population groups at Beaufort-Jasper and Port Wentworth, as well as for other diffuse population groups that make use of the Savannah River; the majority of this dose is due to the drinking water pathway.

For conservatism, as well as demonstrating compliance with DOE Order 5400.5 (100 millirem annual dose limit to an individual from all pathways), the incremental airborne and liquid doses associated with the F-Canyon processing were summed together even though two distinct individuals are assumed to receive a maximum airborne and a maximum liquid dose. In addition, for analysis purposes, it was assumed that tritium would be released to the atmosphere in oxide form. The public impacts from radiological liquid effluent were estimated based on the results provided in the SRS's Interim Management of Nuclear Materials EIS (DOE 1995). This is consistent with the approach used in the recent SRS Spent Nuclear Fuel Management Final EIS (DOE 2000), which used "per unit" values (per metric tons of fuel processed) to estimate liquid doses associated with the PUREX processing of 20 metric tons of heavy metal of declad blanket spent nuclear fuel. This EIS uses the same approach to estimate the radiological doses to the public from potential radiological liquid effluent from PUREX processing.

Table E–12 Annual and Total Radiological Impacts to the Public From Normal Operational Releases During PUREX Processing Under Alternative 3 at SRS

Population [a]				*Maximally Exposed Offsite Individual* [a]			
Air Dose (person-rem)	*Liquid Dose* [b] *(person-rem)*	*Total Collective Dose (person-rem)*	*Latent Cancer Fatalities*	*Air Dose (millirem)*	*Liquid Dose* [b] *(millirem)*	*Total Dose (millirem)*	*Latent Cancer Fatality Risk*
0.019	0.00068	0.020	0.000010	0.00039	0.00012	0.00051	2.6×10^{-10}

[a] The dose values presented apply to both annual and project total, since the processing is done in less than a year.
[b] The dose values were estimated based on the results for processing a similar fuel presented in the Interim Management of Nuclear Materials EIS (DOE 1995).

Worker Impacts

Worker population and worker doses associated with PUREX processing at SRS were based on 300 workers and the site administration dose limit of 500 millirem per year for each worker and are consistent with those presented in the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000). The SRS radiation control program incorporates the DOE Administrative Control Level of 2,000 millirem per year per person established for all DOE activities in DOE Order N441.1. Doses and associated impacts are based on a six-month

processing period. **Table E–13** presents estimated values to both the average worker and entire work force population.

Table E–13 Annual and Cumulative Worker Radiological Impacts from Normal Operational Activities Under Alternative 3 at SRS

Worker Population		*Individual Worker*	
Collective Dose (person-rem per year)	*Latent Cancer Fatalities From Six Months of Processing*	*Individual Dose (millirem per year)*	*Latent Cancer Fatality Risk From Six Months of Processing*
75 [a]	0.015	500 [a,b]	1.0×10^{-4}

[a] Processing of blanket spent nuclear fuel will require six months of F-Canyon operation, yielding half of the annual doses presented.
[b] 500 millirem is an annual ALARA administrative dose limit at SRS. The average worker dose is about 50 millirem (DOE 2000).

E.4.4 SRS Building 105-L Melt and Dilute Radiological Releases and Impacts (Alternative 5)

Melt and dilute processing at Building 105-L would release radioactive gaseous fission products during treatment of about 57 metric tons of heavy metal of EBR-II and Fermi-1 blanket spent nuclear fuel. Since declad and cleaned blanket spent nuclear fuel would be packaged and sent to SRS, no additional gaseous fission products would be expected to be present in that fuel. However, it was assumed conservatively that the gaseous fission products in the blanket spent nuclear fuel would remain within the fuel matrix and would be released to the environment from the facility stack (62 meters [203 feet] high) during melt and dilute processing at SRS. As a result, there would be incurred doses to the public associated with these operations. The duration of the melt and dilute process was estimated to be about three years, based on the current design throughput of the melter and an assumption that the final metallic high-level radioactive waste product from this process would contain about 30 percent depleted uranium in aluminum alloy (WSRC 1999).

Gaseous Emissions

Based on the assumption that the entire fission gas inventory would remain within the fuel matrix after the decladding and cleaning process, it was assumed the inventory of krypton-85 and tritium (H-3) would be released during the melt and dilute process. Using the gaseous fission product inventory provided in Appendix D for the EBR-II and Fermi-1 blanket spent nuclear fuel, the potential airborne radiological release quantities were estimated and are presented in **Table E–14**. These inventories then were used to calculate the population doses from air emissions.

Table E–14 Annual Radiological Releases During Normal Melt and Dilute Operations at Building 105-L Under Alternative 5 at SRS

Isotope	*Releases [a] to Air (curies)*
H-3	54
Kr-85	396

[a] There are no liquid releases associated with melt and dilute processing at SRS.

Liquid Effluent

The melt and dilute process would not produce liquid effluent.

Population Impacts

Estimated annual radiological impacts associated with melt and dilute operations at SRS for the maximally exposed offsite individual and the general population residing within the 80 kilometer (50 mile) radius surrounding Building 105-L are presented in **Table E–15**. For analysis purposes, the released tritium was assumed to be in oxide form.

Table E–15 Annual Radiological Impacts to the Public From Normal Operational Releases During Melt and Dilute Processing at Building 105-L Under Alternative 5 at SRS

Population		*Maximally Exposed Offsite Individual*	
Collective Dose (person-rem)	*Latent Cancer Fatalities*	*Dose (millirem)*	*Latent Cancer Fatality Risk*
0.0076	3.8×10^{-6}	0.00010	5.0×10^{-11}

Worker Impacts

Worker population and worker impact doses associated with melt and dilute processing at SRS were based on 100 workers and the site administrative dose limit of 500 millirem per year for each worker and are consistent with those presented in the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000). The SRS radiation control program incorporates the DOE Administrative Control Level of 2,000 millirem per year per person established for all DOE activities in DOE Order N441.1. Doses and associated impacts are based on a three-year processing period. **Table E–16** presents estimated values to both the average worker and entire work force population.

Table E–16 Annual and Cumulative Worker Radiological Impacts From Normal Operational Activities During Melt and Dilute Operations at Building 105-L Under Alternative 5 at SRS

Worker Population		*Individual Worker*	
Collective Dose (person-rem per year)	*Latent Cancer Fatalities From Three Years Melt and Dilute Processing*	*Individual Dose (millirem per year)*	*Latent Cancer Fatality Risk From Three Years Melt and Dilute Processing*
50	0.060	500 [a]	0.00060

[a] 500 millirem per year is the site annual ALARA administrative dose limit at SRS. The average worker dose is about 50 millirem per year (DOE 2000).

E.4.5 Melt and Dilute Processing at ANL-W (Alternative 6)

In Alternative 6, the blanket and driver spent nuclear fuel elements would need to be prepared at the ANL-W facilities prior to their processing at ANL-W. Preparation of the fuel at ANL-W for the melt and dilute process requires only that the fuel be cleaned to remove sodium prior to melt and dilute processing; decladding of the blanket and driver spent nuclear fuel is not necessary. This activity would be performed in the argon cell of the Hot Fuel Examination Facility. The following discussion addresses the radiological impacts of normal operations at ANL-W for the preparation and melt and dilute treatment of the blanket and driver spent nuclear fuel.

Gaseous Emissions

Fuel preparation would occur at the Hot Fuel Examination Facility. These activities would cause gaseous fission products to be released into the argon cell. As stated earlier in Section E.4.1, krypton-85 and elemental tritium (H-3) are the most prevalent gaseous radionuclides that would be released to the environment. The

tritium released into the cell would not be oxidized because of a very low presence of oxygen and humidity in the argon cell. The argon cell also contains an equilibrium concentration of other radionuclide isotopes. Section E.4.1 provides a list of various isotopes that are present in the argon cell in nanocuries (10^{-9} curies) and are released to the atmosphere through the facility stack, along with the krypton-85 and elemental tritium.

The melt and dilute alternative consists of two distinct operations. The spent nuclear fuel first would be declad and cleaned to remove as much sodium as possible and then would be treated using melt and dilute process. Decladding and cleaning operations may start as early as 2003 and could continue for six years. The melt and dilute treatment would be expected to begin two years after the start of the decladding and cleaning operations. The gaseous fission products (including tritium and krypton) would be released during the decladding and cleaning of the spent nuclear fuel, when the fuel temperature would be raised to approximately 500 °C (930 °F). During the melt and dilute process itself, additional radionuclides would be volatilized and particulates would be released. The volatilized elements would be condensed and collected while the airborne particulates would be filtered through a filtration system that reduces any release by a factor of at least 0.00001. Analysis performed for the evaluation of normal operations for Alternative 1, Section E.4.1, showed that, for similar conditions, over 99 percent of the population and maximally exposed offsite individual doses would come from the release of tritium and krypton from processing both the blanket and driver spent nuclear fuel. Therefore, the doses from the release of tritium and krypton are used to represent the offsite impacts of normal operation releases. These releases would occur during the six years of decladding and cleaning activities, beginning in approximately 2003.

The maximum released curies of radioactive gaseous emissions would occur when preparation of the blanket and driver spent nuclear fuel is performed simultaneously. This simultaneous operation was estimated to occur over a six-year period starting in 2003. Based on a blanket spent nuclear fuel preparation throughput of 10 metric tons of heavy metal and a driver spent nuclear fuel process rate of about 1.7 metric tons of heavy metal annually, about 2,162 curies of elemental tritium and 32,650 curies of krypton-85 would be released to the atmosphere annually (see **Table E-17**). This release rate would last about two years, or until all of the EBR-II blanket spent nuclear fuel and the driver spent nuclear fuel assemblies were processed. Afterward the release rate would drop during the processing of the Fermi-1 blanket spent nuclear fuel (the release rate for the processing of 10 metric tons of heavy metal of Fermi-1 blanket spent nuclear fuel is less than 1 curie of elemental tritium and 6.6 curies of krypton-85).

Table E–17 Maximum Annual Radiological Gaseous Emissions During Melt and Dilute Operations Under Alternative 6 at ANL-W

Spent Nuclear Fuel Type	*Facility*	*Maximum Processing Rate (metric tons of heavy metal per year)*	*Duration (years)*	*Annual Release (curies)*	
				Tritium	*Krypton-85*
Driver fuel	Hot Fuel Examination Facility	1.7	2	2091	32,130
EBR-II blanket fuel	Hot Fuel Examination Facility	10	2.4	71.2	520
Fermi-1 blanket fuel	Hot Fuel Examination Facility	10	3.6	0.76	6.6
Maximum annual release [a]			2	2,162	32,650

[a] Maximum annual release rate applies to the two years during which both EBR-II driver and blanket spent nuclear fuel are processed.

Population Impacts

The maximum annual doses to the maximally exposed offsite individual and the general public residing within the 80 kilometer (50 mile) radius surrounding ANL-W are presented in **Table E–18**. As stated in Section E.4.1, the dose resulting from the release of tritium is highly dependent upon its chemical form. The doses presented in Table E–18 result from releases that are assumed to be 1 percent oxidized tritium, the same

assumption used in the analysis of Alternative 1. These impacts are calculated for the preparation and processing of the sodium-bonded blanket and driver spent nuclear fuel assemblies and the total maximum impacts. During the four-year period when only EBR-II or Fermi-1 blanket spent nuclear fuel is being processed, the doses and latent cancer fatality risk would be smaller than the total presented in the table, reduced in direct proportion to the amount of material released.

Table E–18 Annual and Total Radiological Impacts to the Public From Operational Releases Under Alternative 6 at ANL-W

		Population		*Maximally Exposed Offsite Individual*	
Year(s) of Processing	*Spent Nuclear Fuel Type*	*Collective Dose (person-rem per year)*	*Latent Cancer Fatalities (number of cancers)*	*Dose (millirem per year)*	*Latent Cancer Fatality Risk*
1 to 2	Driver	0.012	6.0×10^{-6}	0.0020	1.0×10^{-9}
	EBR-II blanket	0.00028	1.4×10^{-7}	0.000048	2.4×10^{-11}
	All fuel	0.012	6.1×10^{-6}	0.0020	1.0×10^{-9}
3	EBR-II blanket	0.00011	5.5×10^{-8}	0.000019	9.5×10^{-12}
	Fermi-1 blanket	1.9×10^{-6}	9.5×10^{-10}	3.3×10^{-7}	1.6×10^{-13}
	All fuel	0.00012	5.6×10^{-8}	0.000019	9.7×10^{-12}
4 to 6	Fermi-1 blanket	3.2×10^{-6}	1.6×10^{-9}	5.5×10^{-7}	2.7×10^{-13}
	All fuel	3.2×10^{-6}	1.6×10^{-9}	5.5×10^{-7}	2.7×10^{-13}
Project total [a]		0.024	0.000012	0.0040	2.0×10^{-9}

[a] Maximum annual radiological impacts occur during concurrent processing of EBR-II driver and blanket spent nuclear fuel.

It should be noted that the radiological impacts presented in Table E–18 are based on the assumption of simultaneous operation of blanket and driver fuel. If the fuel preparation were to be performed along with the melt and dilute process for each fuel type separately, then the emissions would occur over a 10-year period starting in 2005. This would result in a lower annual dose to the public over a longer duration leading to the same project total dose as presented in Table E–18.

Worker Impacts:

Due to the uncertainties in the start of operation, for the purposes of analysis in this EIS, DOE assumes that the fuel preparation would start at 2003 and the melt and dilute processing of all fuel would end in 2015, for a total of 12 years of operations. The worker activities during fuel preparation and melt and dilute process would be similar to the activities for Alternatives 1 through 5 at ANL-W. Therefore, the annual worker population dose and average worker dose would be 22 person-rem and 60 millirem, respectively. If these estimates were extended over 12 years of treatment activities plus one year for the deactivation of the facility (with 33-person-rem of dose), the project total worker population dose would be 297 person-rem, leading to a risk of 0.12 latent cancer fatalities.

E.4.6 Storage/Direct Disposal (No Action Alternative)

In the No Action Alternative, the blanket and driver spent nuclear fuel assemblies would remain in their current storage facilities at ANL-W and Idaho Nuclear Technology and Engineering Center (INTEC) until a future disposal option is selected. Potentially, the spent nuclear fuel could remain in its current location until its preparation for disposition sometime before the end of 2035. (All of the sodium-bonded spent nuclear fuel must be removed from the site and moved out of the State of Idaho to fulfill the requirements of the DOE-State of Idaho Settlement Agreement and Consent Order [see Appendix K].) The only activities associated with continued storage, other than monitoring, would be the repackaging of 5 metric tons of blanket spent nuclear

fuel (over a two-year period ending in 2001); the repackaging of spent nuclear fuel found to have degraded (e.g., leaking fuel and storage canister); the transfer of 1.2 metric tons of driver spent nuclear fuel currently in wet storage at Basin 603 to dry storage; and the repackaging of all of the spent nuclear fuel prior to its removal from the State of Idaho. The following discussion addresses the radiological impacts of normal operations at ANL-W and INTEC for the continued storage of the blanket and driver spent nuclear fuel.

Gaseous Emissions:

Under both options in this alternative potential radiological releases from sodium-bonded spent nuclear fuel are very small. Under either option, the sodium-bonded spent fuel would remain in storage in sealed canisters while at INEEL (i.e., INTEC or ANL-W). This fuel needs to be removed from the State of Idaho by January 1, 2035, consistent with the DOE-State of Idaho Settlement Agreement and Consent Order, (see Appendix K). However, degradation of sodium-bonded spent nuclear fuel or its enclosure (e.g., a sealed canister) during storage cannot be ruled out. It is expected that a percentage (a small fraction) of the fuel would be degraded during storage, allowing its gaseous fission products to enter the storage canister. These fission gases would be released to the environment only if the sealed canister were to fail or be opened during fuel handling for examination and repackaging.

The current experience of sodium-bonded spent nuclear fuel storage at INTEC (Basins 603 and 666, wet storage facilities) and ANL-W (Radioactive Scrap and Waste Facility, a dry storage facility) indicates some small fuel degradation problems during the storage period (ANL 2000). For example, during the Electrometallurgical Treatment Research and Demonstration Project, only one canister was observed to have a degraded driver fuel element among the 0.4 metric tons of heavy metal (100 assemblies) of EBR-II driver spent nuclear fuel treated. The degraded fuel was among 6,100 fuel elements that were in dry storage for an average of about four years. Based on this limited experience, the likelihood of fuel degradation during dry storage for the driver spent nuclear fuel would be about 0.005 percent per year. All fuel stored at the Radioactive Scrap and Waste Facility is in cathodically protected liners. At ANL-W, no failures of cathodically protected liners have occurred; therefore, any fuel degradation while in storage is not expected to lead to immediate releases into the atmosphere. Since the driver spent nuclear fuel that failed was in storage for a short period, the failure rate was adjusted to 0.015 percent per year, considering an error factor of 3 (or an uncertainty factor of 10). Therefore, if the fuel were to remain in dry storage for 35 years, about 0.5 percent of the sodium-bonded driver spent nuclear fuel would be in a degraded condition.

The EBR-II fuel at INTEC's Basins 603 and 666 are stored inside stainless steel sealed cans to prevent the contact of basin water with the fuel cladding. The experience at INTEC indicates a higher likelihood of fuel/can degradation in wet storage. A total of 3,624 fuel cans of spent nuclear driver fuel currently is stored at INTEC's Basins 603 and 666. There were 2,148 cans in Basin 603, with an average storage of about 17 years. During this period, 10 cans have shown degradation and water in-leakage, leading to an estimated fuel can failure rate of about 0.03 percent per year. This failure rate was adjusted to 0.10 percent per year, consistent with the assumption made for the driver spent nuclear fuel. Water in-leakage had caused fuel degradation and hydrogen generation from sodium water reactions. The sodium-bonded spent nuclear fuel in Basin 666 has been in storage for about 12 years on average with no observed fuel can failure. All spent nuclear fuel, including the sodium-bonded spent nuclear fuel currently in Basin 603 at INTEC, is to be transferred to dry storage by December 2000, independent of the actions considered in this EIS. During transfer, each fuel can containing sodium-bonded spent nuclear fuel would be examined for water in-leakage. If a fuel can were found to be degraded (containing water), it would be packaged and sent to ANL-W for further examination and repackaging. After transfer to a dry storage facility at INTEC, the likelihood of fuel degradation would be similar to that at ANL-W. The sodium-bonded spent nuclear fuel in Basin 666 would remain in (wet) storage until the planned defueling and facility closure in the year 2023.

In the basin (wet storage), fuel can degradation would cause a fuel-water reaction, producing hydrogen gas, which would create bubbles in the basin leading to failure detection. Upon detection, the fuel can would be removed and sent to ANL-W for further examination and repackaging. If no action were taken to treat the sodium-bonded spent nuclear fuel by 2023, the fuel would be removed from the basin and placed in storage or repackaged in preparation for removal from the State of Idaho by 2035.

Based on the above experience, the likelihood of fuel failure during dry storage was estimated to be about 0.015 percent per year. When the fuel is in dry storage, fuel degradation would not be detected. Over a storage period of up to 2035, it was estimated conservatively that about 0.5 percent of the fuel would be in a degraded condition. This estimate also would be used for the blanket spent nuclear fuel, even though no blanket spent nuclear fuel element failures during storage have been observed. The likelihood of fuel/can failure in wet storage is about 0.10 percent per year. Therefore, for consistency with the assumption of driver spent nuclear fuel failure in dry storage, it was assumed that about 3 percent of the spent nuclear fuel would have failed during the wet storage period of up to 2035.

Using the above spent nuclear fuel failure assumptions, the estimated radiological gaseous emissions during each option of the No Action Alternative are summarized below.

- *Continued Storage Option*—Under this option, only 107 cans containing about 5 metric tons of heavy metal of blanket spent nuclear fuel at ANL-W would be repackaged and returned to dry storage within the first two years. This would lead to a release of about 0.04 curies of tritium and 0.3 curies of krypton-85 over the first two years. Over the same period, the releases of other gaseous fission products, such as iodine-129, would be less than 10^{-7} curies. The spent nuclear fuel in INTEC's Basin 666, which was assumed to remain in the pool up to 2035, would release about 1 curie of tritium oxide (the elemental tritium in the fuel was assumed to be oxidized in the water), 15.1 curies of krypton-85, and 1.5×10^{-6} curies of iodine-129 annually. At some future time, all sodium-bonded spent nuclear fuel at ANL-W would have to be repackaged in preparation for transferring out of the State of Idaho by 2035, consistent with the DOE-State of Idaho Settlement Agreement and Consent Order. The spent nuclear fuel in dry storage at ANL-W and INTEC would release 16.91 curies of tritium oxide, 254.1 curies of krypton-85, and 0.000011 of iodine-129 during fuel repackaging for removal which would occur over three years. The total radiological releases over 35 years would be: 50.51 curies of tritium oxides, 760.3 curies of krypton-85, and 0.000018 curies of iodine-129. Due to uncertainties about when the repackaging would occur, the blanket and driver spent nuclear fuel radionuclide inventories were not decayed beyond the 2000 calendar year. This could result in overestimating the gaseous tritium content by a factor of 8.

- *Direct disposal*—Under this option, all the sodium-bonded spent nuclear fuel at INTEC or ANL-W would be repackaged at ANL-W. The activities to repackage the sodium-bonded fuel in high-integrity cans would occur over three years. These activities would occur sometime after those performed in the first two years under the storage option of the No Action Alternative and before January 2035, the target date for removal of spent nuclear fuel from the State of Idaho. The fuel currently at INTEC would be transferred to ANL-W between 2003 and 2023. The 2023 date corresponds to the target date for closure of the facility containing Basin 666. Therefore, similar to the previous option, the releases under this option would occur over two distinct periods: (1) over two years during repackaging of the blanket spent nuclear fuel (see continued storage option above); and (2) over three years during repackaging and preparation for direct disposal. Since similar activities are performed under both options, the total radiological releases also would be similar; that is, about 51 curies of tritium, 760 curies of krypton-85, and 0.00002 curies of iodine-129 over the entire period.

Population Impacts:

The doses to the maximally exposed offsite individual and the general public residing within the 80 kilometer (50 mile) radius surrounding the ANL-W site are presented in **Table E–19.** All releases producing these impacts are modeled as originating from ANL-W. Due to the relative locations of ANL-W, INTEC, and the surrounding population, the impacts of releases from INTEC are bounded by the impacts of releases from ANL-W. The dose resulting from the release of tritium is highly dependent upon its chemical form. The doses in Table E–19 result from releases that are assumed to be oxidized tritium.

Table E–19 presents the radiological impacts for the storage option as described above (impacts from the direct disposal option would be similar), and includes contributions from the following releases:

- The repackaging of 5 metric tons of blanket spent nuclear fuel during the first two years,
- Leakage from 2 metric tons of driver spent nuclear fuel in wet storage for one year,
- Leakage from 0.8 metric tons of driver spent nuclear fuel in wet storage (Basin 666) for 31 years (1.2 metric tons would be moved from wet [Basin 603] to dry storage within the first year), and
- The repackaging of all the stored sodium-bonded spent nuclear fuel during the final three years.

Table E–19 Annual and Total Radiological Impacts to the Public From Normal Operations Under the No Action Alternative

		Population		*Maximally Exposed Offsite Individual*	
Year(s) of Storage	*Spent Nuclear Fuel Type*	*Collective Dose (person-rem per year)*	*Latent Cancer Fatalities (number of cancers)*	*Dose (millirem per year)*	*Latent Cancer Fatality Risk*
1	Driver	0.00063	3.2×10^{-7}	0.00011	5.5×10^{-11}
	Blanket	4.8×10^{-6}	2.4×10^{-9}	8.3×10^{-7}	4.1×10^{-13}
	All fuel	0.00064	3.2×10^{-7}	0.00011	5.6×10^{-11}
2	Driver	0.00025	1.3×10^{-7}	0.000044	2.2×10^{-11}
	Blanket	4.8×10^{-6}	2.4×10^{-9}	8.3×10^{-7}	4.1×10^{-13}
	All fuel	0.00026	1.3×10^{-7}	0.000045	2.3×10^{-11}
3 to 32	Driver	0.00025	1.3×10^{-7}	0.000044	2.2×10^{-11}
	Blanket	0	0	0	0
	All fuel	0.00025	1.3×10^{-7}	0.000044	2.2×10^{-11}
33 to 35	Driver	0.0014	7.0×10^{-7}	0.00024	1.2×10^{-10}
	Blanket	0.000072	3.6×10^{-8}	0.000013	6.3×10^{-12}
	All fuel	0.0015	7.5×10^{-7}	0.00026	1.3×10^{-10}
Project total [a]		0.013	6.5×10^{-6}	0.0023	1.1×10^{-9}

[a] Annual maximum occur during repackaging of spent nuclear fuel in preparation for shipping off site.

Worker Impacts:

The worker activities under the No Action Alternative during spent nuclear fuel repackaging would be similar to some of the activities performed under Alternative 1. Therefore, for the five years that repackaging activities are ongoing, the first two years and the last three years that the fuel remains on site, the annual worker dose and the worker population dose would be bounded by those values provided in Section E.4.1.

E.5 Impacts of Exposures to Hazardous Chemicals on Human Health

The potential impacts of exposure to hazardous chemicals released to the atmosphere were evaluated for routine operations associated with the alternatives analyzed in this EIS. The public residing at the site boundary was the receptor considered in this evaluation. Health impacts to workers from hazardous chemicals were not evaluated quantitatively because of the use of personal protective equipment and engineering process controls. Their exposure is limited to levels within applicable Occupational Safety and Health Administration Permissible Exposure Limits, or the American Conference of Governmental Industrial Hygienists Threshold Limit Values.

Human health effects could result from exposure to hazardous/toxic chemicals through one or more of the three common pathways: inhalation, ingestion, and/or dermal (skin) contact. The effects from a particular pathway will depend essentially on the properties of the toxic chemical, its concentration in one or more environmental media (air, water, and soil), and human behavior. Exposure may be dominated by contacts with chemicals in a single medium or may reflect concurrent contacts with multiple media. Therefore, the exposure assessment provides an estimate of how chemicals travel to a receptor, and how those chemicals come into contact with the receptor's body. It also determines whether the chemicals present in the environmental medium are of sufficient concentration to cause significant adverse effects. The exposure assessment assumes inhalation to be the only pathway and air the only medium. This simplification is based principally on the volatility of the chemicals released. Normal human behavior also is considered (i.e., an individual is assumed to perform activities under normal conditions). To maximize the impact of the exposure, the analysis also assumes that the released chemicals will remain in the air with no or negligible partitioning to other media (i.e., water and ground). So no dermal contact or ingestion is considered in this assessment.

Hazardous chemical releases from routine operations generally are expected to result in concentrations below levels that would cause acute toxic health effects. Acute toxic health effects generally result from short-term exposure to relatively high concentrations of the toxic contaminant, such as those resulting from accidental releases. Long-term exposures to lower concentrations can produce adverse chronic health effects, both carcinogenic and noncarcinogenic. Excess incidences of cancer are the endpoint of carcinogenic effects. However, a spectrum of chemical-specific noncancer health effects (e.g., headaches, skin irritation, neurotoxicity, immunotoxicity, reproductive and genetic toxicity, liver/kidney toxicity, and developmental toxicity) could be observed for noncarcinogenic compounds.

E.5.1 Methodology

This EIS estimates the noncancer health effects by comparing the annual concentrations of contaminants to the Reference Concentrations published in the Integrated Risk Information System (EPA 1999). The potential toxic effects on an individual from exposure to a toxic chemical are evaluated by dividing the estimated inhalation concentration of that chemical by its Reference Concentration value to obtain a noncancer hazard quotient (EPA 1989). For exposure to multiple compounds, hazard quotients are calculated for each toxic chemical and then are summed to generate a hazard index:

$$HI = \sum_i \frac{CA_i}{RfC_i}$$

where

RfC_i = Reference Concentration for chemical *I* (in micrograms per cubic meter)
CA_i = Concentration of the chemical *I* in the air (in micrograms per cubic meter)
HI = Hazard Index

The hazard index is the estimate of the total noncancer toxicity impact. According to the EPA risk assessment guidelines, if the hazard index value is less than or equal to 1, the exposure is unlikely to produce adverse toxic effects. However, if it exceeds 1, adverse toxic effects may result from exposure to the considered chemicals.

The risks from exposure to carcinogenic chemicals are evaluated using chemical-specific unit risk factors, which are the estimates of the upper-bound lifetime probability of an individual developing cancer from exposure to the chemical and the chemical concentration in the air. The unit risk factors for carcinogenic chemicals are provided in the EPA's Integrated Risk Information System database. Therefore, for carcinogenic chemicals, the risk is estimated by the following equation (EPA 1989):

$$\text{Risk} = 1 - \exp[-\text{CA} \times \text{URF}]$$

where

CA	=	Contaminant concentration in the air (in micrograms per cubic meter)
URF	=	Unit risk factor for inhalation specific to the contaminant obtained from the Integrated Risk Information System in units of cancer per micrograms per cubic meter

Since the value in the bracket is generally small (less than 0.01), the equation is simplified to:

$$\text{Risk} = \text{CA} \times \text{URF}$$

E.5.2 Assumptions

The air is assumed to be the principal medium by which an individual would be exposed to released hazardous chemicals, and the health effects are calculated based solely on inhalation pathway. In addition, no synergistic or antagonistic effects are assumed to occur from the exposure to multiple hazardous chemicals.

Cancer risks associated with exposure to carcinogenic chemicals were not summed to provide a single cancer risk value. In terms of risk evaluation, a value integrated over multiple chemicals is not always appropriate. One cannot simply add the risk values of individual chemicals to calculate the overall risk. With the risk assessment guidelines and the weight of evidence (EPA 1999), a new approach to carcinogenic risk characterization is being implemented. Thus, even though several chemicals may be shown to induce cancer, they do not necessarily act on the same organ. For example, benzene and formaldehyde are both carcinogenic. Formaldehyde could induce nasal cancer (Andjelkovitch, et. al. 1995), while benzene could cause leukemia. Thus, their residual cancer risk is not cumulative, and the cancer risk for each carcinogenic chemical would be presented separately.

E.5.3 Hazardous Chemical Releases to the Environment and Associated Impacts

This section summarizes the estimated hazardous chemical releases to the environment as well as resulting impacts associated with various alternatives assessed in this EIS.

E.5.3.1 Hazardous Chemical Impacts at ANL-W (All Alternatives)

Under all alternatives, including No Action, small quantities of hazardous chemicals are generated from the operation of the emergency diesel generators supporting both the Fuel Conditioning Facility and Hot Fuel Examination Facility at ANL-W. The emissions from these are independent of any of the processes addressed in this EIS. The released chemicals include acetaldehyde, acrolein, benzene, butadiene, formaldehyde, and toluene. The emissions from these diesel generators were modeled as a volume source releasing at ground level. In addition, the electrometallurgical treatment of driver spent nuclear fuel under Alternatives 1 through

5 releases small quantities of cadmium. This release would occur as an elevated release (61 meters [200 feet]) from the Fuel Conditioning Facility stack.

Site boundary hazardous chemical concentrations in the atmosphere from releases at ANL-W were estimated using the SCREEN 3 computer program (Version 96043), an EPA-approved worst-case screening model (EPA 1995). The model predicts 1-hour concentrations at the site boundary based on a set of worst-case meteorological conditions. Concentrations were predicted at 16 sectors along the site boundary, assuming a flat terrain. The maximum 1-hour concentration at the site boundary then was selected for the determination of health effects. This concentration was converted to an annual concentration using a regulatory-approved scaling factor of 0.05 (SCDHEC 1993). **Table E–20** summarizes the results. These results indicate that no adverse toxic health effects and cancer potency are expected from exposure to hazardous chemical releases under all alternatives at ANL-W.

Table E–20 Hazardous Chemical Impacts to the Public From Operational Activities at ANL-W for All Alternatives Including No Action

Chemical	*Modeled Emission Rate (grams per second)*	*Annual Concentration (milligrams per cubic meter)*	*Reference Concentration Inhalation (milligrams per cubic meter)*	*Unit Cancer Risk (risk per milligram per cubic meter)*	*Hazard Quotient*	*Cancer Risk*
1,3-Butadiene	0.000012	3.6×10^{-8}	None	0.28	None	9.9×10^{-9}
Acetaldehyde	7.6×10^{-6}	2.3×10^{-8}	0.009	0.0022	2.5×10^{-6}	5.0×10^{-11}
Acrolein	2.4×10^{-6}	7.1×10^{-9}	0.00002	None	0.00035	None
Cadmium	1.5×10^{-10}	3.6×10^{-13}	None	1.8	None	6.5×10^{-13}
Benzene	0.00023	6.9×10^{-7}	None	0.0078	None	5.4×10^{-9}
Formaldehyde	0.000024	7.1×10^{-8}	None	0.013	None	9.2×10^{-10}
Toluene	0.000083	2.5×10^{-7}	0.4	None	6.2×10^{-7}	None
				Hazard Index	0.000353	N/A

N/A = Not applicable.
Source: EPA 1999, modeling results.

E.5.3.2 Hazardous Chemical Impacts at SRS (Alternatives 3 and 5)

Hazardous chemical releases associated with the PUREX and melt and dilute processes at SRS were estimated based on information provided in the SRS Spent Fuel Management Final EIS (DOE 2000). The hazardous chemical release estimates at SRS were essentially independent of the processes evaluated; the chemicals are generated from operation of supporting facilities and equipment (i.e., emergency diesel generator, site-wide powerhouse coal-fired boilers and fuel-oil steam generated boilers). The hazardous chemical release values selected for this EIS were the SRS estimated values that were released during treatment of about 20 metric tons of heavy metal of declad and cleaned EBR-II blanket spent nuclear fuel, similar to the fuel considered for treatment at SRS under the SBSNF EIS. These SRS values were adjusted to account for the mass of spent nuclear fuel being treated (about 57 metric tons of heavy metal) at SRS under Alternatives 3 and 5. In addition, the annual hazardous chemical concentrations were estimated using the 24-hour concentration values given in the SRS EIS and the regulatory-approved scaling factor of 0.125 to convert the 24-hour concentration to an annual concentration (SCDHEC 1993).

Tables E–21 and **E–22** present the results of the hazardous chemical analyses for Alternatives 3 and 5, respectively. These results indicate that no adverse toxic health effects and cancer potency are expected from exposure to hazardous chemical releases under these alternatives at SRS.

Table E–21 Hazardous Chemical Impacts to the Public From Operational Activities Under Alternative 3 at SRS

Chemical	*Annual Concentration (milligrams per cubic meter)* [a]	*Reference Concentration Inhalation (milligrams per cubic meter)*	*Unit Cancer Risk (risk per milligrams per cubic meter)*	*Hazard Quotient*	*Cancer Risk*
Benzene	1.4×10^{-6}	None	0.0078	None	1.1×10^{-8}
Ethyl benzene	1.3×10^{-6}	1	None	1.3×10^{-6}	None
Formaldehyde	1.3×10^{-6}	None	0.013	None	1.6×10^{-8}
Hexane	1.4×10^{-6}	0.2	None	7.1×10^{-6}	None
Manganese	1.3×10^{-6}	0.000050	None	0.025	None
Methyl ethyl ketone	2.5×10^{-6}	1	None	2.5×10^{-6}	None
Methylene chloride	7.1×10^{-7}	None	0.00047	None	3.3×10^{-10}
Naphthalene	1.3×10^{-6}	0.003	None	0.00042	None
Toluene	1.4×10^{-6}	0.4	None	3.5×10^{-6}	None
Vinyl acetate	1.3×10^{-6}	0.2	None	6.3×10^{-6}	None
			Hazard Index	0.025	N/A

N/A = Not applicable.
[a] These concentrations were estimated based on values given in Bickford et al. 1997.
Source: EPA 1999.

Table E–22 Hazardous Chemical Impacts to the Public From Operational Activities Under Alternative 5 at SRS

Chemical	*Annual Concentration (milligrams per cubic meter)* [a]	*Reference Concentration Inhalation (milligrams per cubic meter)*	*Unit Cancer Risk (risk per milligrams per cubic meter)*	*Hazard Quotient*	*Cancer Risk*
Benzene	ND	None	0.0078	None	ND
Ethyl benzene	ND	1	None	ND	None
Formaldehyde	1.3×10^{-6}	None	0.013	None	1.6×10^{-8}
Hexane	1.3×10^{-6}	0.2	None	6.3×10^{-6}	None
Manganese	ND	0.00005	None	ND	None
Methyl ethyl ketone	1.3×10^{-6}	1	None	1.3×10^{-6}	None
Methylene chloride	ND	None	0.00047	None	ND
Naphthalene	1.3×10^{-6}	0.003	None	0.00042	None
Toluene	1.3×10^{-6}	0.4	None	3.1×10^{-6}	None
Vinyl acetate	ND	0.2	None	ND	None
			Hazard Index	0.00043	N/A

N/A = Not applicable, ND = Not detectable.
[a] These concentrations were estimated based on values given in Bickford et al. 1997.
Source: EPA 1999.

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Appendix F

Evaluation of Human Health Effects from Facility Accidents

APPENDIX F
EVALUATION OF HUMAN HEALTH EFFECTS FROM FACILITY ACCIDENTS

F.1 INTRODUCTION

This appendix presents the methodology and assumptions used for estimating potential impacts and risks associated with both radiological and toxic chemical releases, due to postulated accidents, at the facilities being considered for the treatment and management of sodium-bonded spent nuclear fuel. Analysis of radiological impacts is presented in Section F.2. This is followed by a summary of the risk results for the various alternatives. Chemical risk methodologies and results are presented in Section F.3. Information regarding the impacts of normal operations, along with background information on the health impacts from exposure to ionizing radiation, is provided in Appendix E.

F.2 IMPACTS OF RADIOLOGICAL ACCIDENTS ON HUMAN HEALTH

This section addresses the radiological impacts associated with accidents at management facilities. Potential accident scenarios have been identified for both the Argonne National Laboratory-West (ANL-W) and Savannah River Site (SRS) facilities proposed for the treatment and management of sodium-bonded spent nuclear fuel.

F.2.1 Overview of Methodology and Basic Assumptions

For the radiological evaluation, the GENII computer program (PNL 1988) was used to calculate radiation doses to the general population and selected individuals. Appendix E provides the detailed description of this code; therefore, only the GENII data specific to the accident analysis is presented in this appendix.

The impacts of radiation exposure were evaluated for the following population segments for each accident scenario:

- *Noninvolved Worker*—An individual located 100 meters (330 feet) from the radioactive material release point.[1] The dose to the noninvolved worker was calculated for the 50th percentile meteorology only, as specified in the U.S. Department of Energy (DOE) Hazard Categorization and Accident Analysis Techniques Standard (DOE 1992). Noninvolved workers would be exposed unprotected to the plume for a limited time (a maximum of 5 minutes), receiving exposure via inhalation, air immersion, and ground surface pathways only.

- *Maximally Exposed Offsite Individual*—An individual member of the public living at the management site boundary and receiving the maximum exposure. This individual is located directly downwind of the accident and would be exposed to radioactivity via inhalation, ingestion, air immersion, and ground surface pathways. The individual would be exposed to the plume for the entire release duration.

- *Population*—The general public living within an 80-kilometer (50-mile) radius of the facility, residing directly downwind of the accident, and receiving the maximum exposure via inhalation, ingestion, air immersion, and ground surface pathways.

[1] *For elevated release, the worker dose was calculated at a point of maximum dose. The distance at which the maximum dose could occur is frequently greater than 100 meters (330 feet) for an elevated release.*

The doses to the maximally exposed offsite individual and the general public were calculated for the 50th and 95th percentile meteorological conditions. Meteorology specific to ANL-W and SRS was used in the evaluation. Site-specific meteorological data was obtained in the form of a joint frequency distribution in terms of percentage of time that the wind blows in specific directions for the given midpoint (or average) wind speed and atmospheric stability. Accident consequences were calculated for both 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 that produce higher calculated exposures, and is defined as that condition 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.

The following conditions were used in the calculations:

- Meteorological Data
  - Site-specific joint frequency distribution weather data were used to define 50th and 95th percentile meteorological conditions for each processing technology at management sites.
  - Any release through a stack was assumed to occur at an elevated level consistent with the site's effluent emission stack height. The effects of plume rise were not credited in the analysis.
  - Mixing layer height is 1,000 meters (3,280 feet). Airborne materials freely diffuse in the atmosphere near the ground level in what is known as the "mixing depth." A stable layer exists above the mixing depth and restricts vertical diffusion above 1,000 meters (3,280 feet).
  - Wet deposition is zero (it was assumed that no rain occurs to accelerate deposition and reduce the size of the area affected by the release).
  - Dry deposition of the cloud was 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 quantity of deposited radioactive material is proportional to the particle size and deposition velocities (in meters per second). The deposited material contributes to the exposure from ground surface radiation and ingestion.
- Inhalation Data
  - Breathing rate is 330 cubic centimeters per second (0.7 cubic feet per minute) for the worker and the general public at the site boundary and beyond (maximally exposed offsite individual and population) during the passage of the plume; it is 270 cubic centimeters per second (0.57 cubic feet per minute) for the general public during the other times.
  - Exposure during passage of the entire plume was assessed for the maximally exposed offsite individual and the population. Exposure to the noninvolved worker is to a portion of the plume (i.e., the noninvolved worker is exposed to the plume for a limited time) because the worker is assumed to take emergency action.
  - Inhalation exposure factors are based on the International Commission on Radiological Protection, Publication 30 (ICRP 1982).

Exposure time assumptions for maximally exposed offsite individuals, workers, and the general public are provided in **Table F–1** below. Since all accident releases would be to the air (either gaseous or suspended

particulates), drinking water, aquatic food ingestion, and any other pathways that may involve liquid exposure were not examined. Additional information, common to the analysis of normal operation and accident impacts, is presented in Appendix E.

Table F–1 GENII Plume and Soil Contamination Exposure Parameters (Postulated Accidents)

Maximally Exposed Offsite Individual			*General Population*		
Inhalation and External Exposure			*Inhalation and External Exposure*		
Exposure Time (hours)	*Breathing Rate (cubic centimeters per second)*	*Soil Contamination (hours)*	*Exposure Time (hours)*	*Breathing Rate (cubic centimeters per second)*	*Soil Contamination (hours)*
100 percent of release time	330	6,136	100 percent of release time	330	6,136

Source: PNL 1988.

Radiological impacts to noninvolved workers from postulated accident scenarios were evaluated at onsite locations where a given incident would cause the highest dose. The noninvolved worker was assumed to have an inhalation exposure time of 5 minutes and an external exposure time to soil contamination of 20 minutes. For a ground-level release accident, a noninvolved worker was assumed to be 100 meters (330 feet) from a given release point; for an elevated release, the worker was situated between 200 and 500 meters (660 and 1,640 feet), depending on the given site's atmospheric dispersion characteristics. All doses to noninvolved workers include a component associated with the intake of radioactivity into the body and another component resulting from external exposure to direct radiation.

The radiation doses to individuals and the public resulting from exposure to radioactive releases were calculated using the following potential pathways:

- *Air immersion*—External direct exposure from immersion in the airborne radioactive material
- *Ground surface*—External direct exposure from radioactive material deposited on the ground
- *Inhalation*—Internal exposure from inhalation of radioactive aerosols and suspended particles
- *Ingestion*—Internal exposure from ingestion of contaminated terrestrial food or animal products

The radiation doses were estimated by the GENII computer program, which uses the dose models recommended by the International Commission on Radiological Protection in Publications 26 and 30 (ICRP 1977, ICRP 1982). Committed dose equivalents[2] are calculated individually for organs such as the gonads, breast, red bone marrow, lungs, thyroid, and bone surface; calculations are combined for the liver, upper large intestine, lower large intestine, small intestine, and stomach. Weighting factors are used for various body organs to calculate weighted or committed effective dose equivalents from radiation inside the body due to inhalation or ingestion. The committed effective dose equivalent value is the sum of the committed dose equivalent to a specific organ weighted by the relative risk to that organ compared to an equivalent whole-body exposure. The deep-dose equivalent for the external exposure pathways (immersion in the radioactive material and exposure to the ground contamination) and the 50-year committed effective dose equivalent for the internal exposure pathways were calculated. The sum of the deep-dose equivalent for external pathways and the committed effective dose equivalent for internal pathways is called the "total effective dose equivalent," or simply, the "total dose" in this environmental impact statement (EIS).

[2]*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."*

The exposure from ingestion of contaminated terrestrial food or animal products is calculated on a yearly basis. It is expected that continued consumption of contaminated food products by the public would be suspended if the projected dose should exceed that of the protective action guidelines in a radiological accident event (EPA 1991). No reduction of exposure because of protective actions or evacuation of the public was accounted for in this analysis, however. This conservative approach may result in overestimating health effects within an exposed population, but allows for consistent comparison between alternatives.

F.2.2 Selection of Facility Accidents for Detailed Evaluations

The alternatives for the treatment of sodium-bonded spent nuclear fuel assume the use of facilities currently in operation, although modifications to SRS Building 105-L would be necessary before it could be used for the melt and dilute alternative. The selection of accident scenarios was based on those evaluated in the safety analysis reports for the facilities.

Postulated facility accident scenarios were developed based on the review of the analyzed accidents in previous safety analysis, risk assessment, and environmental assessment documents at ANL-W and SRS, where the sodium-bonded spent nuclear fuel may be handled or processed. These documents include the following:

- *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement* (DOE 1995a)
- *Environmental Assessment Electrometallurgical Treatment Research and Demonstration Project in the Fuel Conditioning Facility at ANL-West* (DOE 1996a)
- *Fuel Cycle Facility Final Safety Analysis Report* (ANL 1998a)
- *Safety Analysis Report for the Hot Fuel Examination Facility* (ANL 1998b)
- *Accident Assessments for Idaho National Engineering Laboratory Facilities* (Slaughterbeck et al. 1995)
- *Safety Analysis-200 Area, Savannah River Site F-Canyon Operation, F-Canyon SAR Addendum* (WSRC 1994)
- *Savannah River Site Spent Nuclear Fuel Management Final Environmental Impact Statement* (DOE 2000)

Based on this review of analyzed accident scenarios at ANL-W and SRS facilities that deal with sodium-bonded spent nuclear fuel, a spectrum of potential accidents was identified. This process started with systematically identifying initiating events, subsequent accident progressions, and onsite or offsite releases. Then, based on accident initiators, selected accidents were grouped into the following three categories:

- Natural phenomena (e.g., earthquake, tornado)
- External events (e.g., aircraft crash)
- Process-related events (e.g., explosion, nuclear criticality, fire, spills)

The potential process-related events were further subdivided based on the impact the accident would have on the accident release factors. High-energy events would be expected to damage some of the confinement barriers provided in the facility design and would result in release factors that approach unity. Medium-energy events could reduce the effectiveness of the barriers, but would not be expected to defeat them, while low-energy events would have almost no impact on the ability of the confinement barriers to perform their function.

A review of the accident scenarios indicated that only severe accident conditions (e.g., accidents involving confinement failure) could result in a significant release of radioactive material to the environment or an increase in radiation levels. These severe accident conditions are associated with beyond-design-basis events, combinations of events for which the facility was not specifically designed. While these events could have consequences larger than those associated with design-basis events, their frequency is expected to be much lower than the design-basis event frequency. Natural phenomena (e.g., earthquake) and fire accidents creating a direct path for releases to the environment represent the situation with the most consequences to the public. Some types of accidents, such as procedure violations, spills of small quantities of material containing radioactive particles, and most other types of common human error occur more frequently than the more severe accidents analyzed. However, these accidents do not involve enough radioactive material or radiation to result in significant release to the environment, although the impact to operational personnel may be as significant as that resulting from beyond-design-basis events. The airborne particles from a process-related accident would normally pass through at least one bank and possibly two to four banks of high-efficiency particulate air filters before entering the environment. Spent nuclear fuel handling operations are performed inside such confinement barriers as hot cells or canyon walls. The hot cells are equipped with significant safety features, such as an inert gas atmosphere, pressure control, and heat detection. These features are credited when their operability is not compromised by the sequence of events associated with the accident progression.

While severe accidents (also referred to as beyond-design-basis events) are expected to have the most significant impact, that is, the highest consequences, on the population, these accidents may not have as significant a risk impact on all receptors as higher-frequency, lower-consequence accidents. For this reason, higher-frequency accident scenarios were included in the accident analysis. Three categories of accidents were identified, and at least one accident scenario for each category was selected for analysis. The three categories consist of abnormal events (defined as events with a frequency of greater than 0.001 per year), design-basis events (frequencies between 1×10^{-3} and 1×10^{-6} per year), and beyond-design-basis events (frequencies less than 1×10^{-6}, but limited to those greater than 1×10^{-7} per year).

Based on review of the existing facility analyses and on guidance provided by the U.S. Department of Energy (DOE) in Section 6.9 of *Recommendations for the Preparation of Environmental Assessments and Environmental Impact Statements* (DOE 1993a), the following types of accidents were selected for each processing technology:

- Explosions
- Nuclear criticality
- Fire
- Earthquake
- Aircraft crash
- Spills/drops

Finally, no specific analyses of the results of terrorist or sabotage acts were considered. This is because the existing security measures in effect at the management sites would essentially preclude any sabotage or terrorist activity. In addition, any acts of terrorism would be expected to result in consequences that would be bounded by the results of the accident scenarios selected for detailed evaluation.

F.2.2.1 Accident Scenario Descriptions and Source Terms

This section describes the accident scenarios and corresponding source terms developed for ANL-W and SRS. The spectrum of accidents described below was used to determine the incremental consequences (public and worker doses) and risks associated with the treatment of sodium-bonded spent nuclear fuel at each site. These accident scenarios are consistent with those evaluated in either the facility safety analysis report, facility/site environmental reports, or various related DOE safety documents. Secondary accidents were considered when

identified in the safety documents. The selected documents were identified and referenced in each of the accident scenarios described. When information was required to further clarify the accident condition, update some of the parameters, or facilitate the evaluation process, additional assumptions were made. Sometimes it was necessary to use different assumptions than those used in the referenced report; these are identified also. For example, under the proposed action of this EIS, the material at risk during an earthquake can be different than the materials considered in the facility safety analysis report. This change in assumption is necessary because the evaluations in this EIS focus only on the risk resulting from the implementation of alternatives (an incremental risk) and, therefore, address only the risk associated with the treatment of the sodium-bonded spent nuclear fuel. Cumulative risks can be determined by adding the incremental risks to the existing risks.

F.2.2.1.1 Source Terms

The source term (or building source term) is the amount of respirable radioactive material that is released to the air, in terms of curies or grams, assuming the occurrence of a postulated accident. The airborne source term is typically estimated by the following five-component linear equation:

$$\text{Source term} = \text{MAR} \times \text{DR} \times \text{ARF} \times \text{RF} \times \text{LPF}$$

where:

MAR	=	Material at Risk (grams or curies)
DR	=	Damage Ratio
ARF	=	Airborne Release Fraction (or Airborne Release Rate for continuous release)
RF	=	Respirable Fraction
LPF	=	Leak Path Factor

- *Material at Risk*—The material at risk is the amount of radionuclides (in curies of activity or grams for each radionuclide) available for release when acted upon by a given physical stress (i.e., an accident). The material at risk 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.

- *Damage Ratio*—This is the fraction of material exposed to the effects of the energy, force, or stress generated by the postulated event. For the accident scenarios discussed in this EIS, the value of the damage ratio varies from 0.0001 to 1.

- *Airborne Release Fraction*—This is the fraction of material that becomes airborne due to the accident. In this analysis, airborne release fraction values from the DOE Handbook on airborne release fractions are used (DOE 1994b).

- *Respirable Fraction*—This is the fraction of the material with a 10-micrometer (micron) or less aerodynamic-equivalent diameter particle size that could be retained in the respiratory system following inhalation. The respirable fraction values also are taken from the DOE Handbook on airborne release fractions (DOE 1994b).

- *Leak Path Factor*—The leak path factor accounts for the action of removal mechanisms (e.g., containment systems, filtration, deposition) to reduce the amount of airborne radioactivity that is ultimately released to occupied spaces in the facility or the environment. A leak path factor of 1 (i.e., no reduction) is assigned in accident scenarios involving a major failure of confinement barriers.

F.2.2.1.2 Accident Scenario Descriptions and Source Terms at ANL-W

Accident Scenario Descriptions for Electrometallurgical Treatment Processing—The electrometallurgical treatment process would occur at the Fuel Conditioning Facility and the Hot Fuel Examination Facility at the ANL-W site. This process is detailed in Appendix C. The accident scenarios, identified in **Table F–2** and defined in the following paragraphs, are applicable to the electrometallurgical treatment process as proposed at ANL-W. This section also provides information addressing the material at risk and the various release fractions used to determine the source term for each accident selected for analysis.

Table F–2 Selected Accident Scenarios for Electrometallurgical Treatment Processing at ANL-W

Scenario	*Frequency (per year)*
Process-related spills/drops	
a. Salt powder spill	0.01
b. Cask drop	0.01
c. Salt transfer drop	1×10^{-7}
Transuranic waste fire	0.001
Explosion	Not applicable
Design-basis earthquake	0.0002 [a] / 0.008 [b]
Aircraft crash	6×10^{-7} to 1×10^{-8}
Nuclear criticality	Less than 10^{-7}
Beyond-design-basis earthquake	0.00001

[a] At the Fuel Conditioning Facility.
[b] At the Hot Fuel Examination Facility.

Each accident scenario description sets the condition of the accident and provides a summary of material involved. As stated earlier, some of these accident scenarios are generic, but their applications are consistent with those evaluated in various ANL-W environmental and safety analyses. These scenarios include process-specific as well as generic and process-independent accidents. Tables F–3 through F–8 provide summaries of the accidents analyzed, the material at risk, and the release factors based on the fuel type expected to produce the most significant consequences, typically either Experimental Breeder Reactor-II (EBR-II) blanket or driver spent nuclear fuel, for each postulated accident.

- *Operational accident causing a salt powder spill in the Hot Fuel Examination Facility Main Cell*—Solidified electrorefiner salt is sent from the Fuel Conditioning Facility to the Hot Fuel Examination Facility for processing into a final ceramic waste form. It is brought into the Hot Fuel Examination Facility in solid form and ground. The grinder is located in the Main Cell on a raised floor. In this accident scenario, it was assumed that during a transfer operation, the contents of a ground salt container are spilled into the pit beneath the floor. A portion of the salt powder becomes airborne and is carried through the ventilation system to the high-efficiency particulate air filters and released through the building stack. The release was assumed to occur over a one-hour period. The frequency of this accident was set at 0.01 per year, based on the Safety Analysis Report for the Hot Fuel Examination Facility (ANL 1998b).

  To estimate the fission product inventory in the electrorefiner salt, the option of not blending fuel types during electrorefining was used. The salt was assumed to come from the treatment of 5.56 metric tons of heavy metal of EBR-II blanket spent nuclear fuel elements (Goff et al. 1999b) or 1.1 metric tons of heavy metal of EBR-II driver spent nuclear fuel elements (Goff et al. 1999a), the point at which bulk replacement of salt in the electrorefiner is required either when the sodium limit is reached or when the treatment of each fuel type is completed. For the fuel types selected to represent the driver and blanket spent nuclear fuel, the fission product inventory in the salt would be conservative. Based on the Safety Analysis Report for the Hot Fuel Examination Facility (ANL 1998b), the material at risk was assumed to be 100 kilograms of ground salt containing the radionuclide concentrations as shown in **Table F–3**. Radionuclide distributions were

developed for both EBR-II driver and blanket spent nuclear fuel. The radionuclide distributions for driver and blanket spent nuclear fuel are based on an average plutonium concentration in electrorefiner salt of 1.76 and 7.98 percent by weight, respectively (Goff et al. 1999a and 1999b). Portions of the spilled salt would become airborne. The maximum measured value for the 3-meter (10-foot) free-fall of dry cohesionless particles, with a mass median diameter of 1 to 2 microns, results in an airborne release fraction of 0.002 and a respirable fraction of 0.3 (DOE 1994b). The median particle size of the salt after grinding is approximately 200 microns, with only about 1 percent less than 20 microns in diameter (ANL 1999). The analysis, therefore, conservatively assumed that about 1 percent of the ground salt would have characteristics capable of resulting in the airborne release and respirable fractions identified above, resulting in a damage ratio of 0.01. The ventilation system and high-efficiency particulate air filters were assumed to function normally. The ventilation system consists of a two-stage high-efficiency particulate air filtration system were equivalent, with a first-stage high-efficiency particulate air filter efficiency of 99.9 percent and a second stage efficiency of 99 percent. Therefore, the leak path factor through the high-efficiency particulate air filters is 0.00001.

Table F–3 Material at Risk and Release Fraction Values for a Salt Powder Spill Accident at ANL-W

Material at Risk [a]							*Source Term (curies)*	
Isotope	*Blanket (curies)*	*Driver* [b] *(curies)*	*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Blanket*	*Driver*
Sr-90	580	35,000	0.01	0.002	0.3	0.00001	3.48×10^{-8}	2.10×10^{-6}
Y-90	580	35,000	0.01	0.002	0.3	0.00001	3.48×10^{-8}	2.10×10^{-6}
I-129	0.00104	0.0131	0.01	0.002	0.3	0.00001	6.24×10^{-14}	7.86×10^{-13}
Cs-134	9.63	313	0.01	0.002	0.3	0.00001	5.78×10^{-10}	1.88×10^{-8}
Cs-137	1,240	39,200	0.01	0.002	0.3	0.00001	7.44×10^{-8}	2.35×10^{-6}
Ba-137M	1,180	37,100	0.01	0.002	0.3	0.00001	7.08×10^{-8}	2.23×10^{-6}
Ce-144	45.1	526	0.01	0.002	0.3	0.00001	2.71×10^{-9}	3.16×10^{-8}
Pr-144	45.1	526	0.01	0.002	0.3	0.00001	2.71×10^{-9}	3.16×10^{-8}
Pm-147	292	14,700	0.01	0.002	0.3	0.00001	1.75×10^{-8}	8.82×10^{-7}
Sm-151	71.9	948	0.01	0.002	0.3	0.00001	4.31×10^{-9}	5.69×10^{-8}
Eu-154	5.28	101	0.01	0.002	0.3	0.00001	3.17×10^{-10}	6.06×10^{-9}
Eu-155	34.6	677	0.01	0.002	0.3	0.00001	2.08×10^{-9}	4.06×10^{-8}
Th-228	0.000111	0.0091	0.01	0.002	0.3	0.00001	6.66×10^{-15}	5.48×10^{-13}
Np-237	0.00602	0.0513	0.01	0.002	0.3	0.00001	3.61×10^{-13}	3.08×10^{-12}
Pu-238	6.44	66.8	0.01	0.002	0.3	0.00001	3.86×10^{-10}	4.01×10^{-9}
Pu-239	517	108	0.01	0.002	0.3	0.00001	3.10×10^{-8}	6.48×10^{-9}
Pu-240	35.5	3.67	0.01	0.002	0.3	0.00001	2.13×10^{-9}	2.20×10^{-10}
Pu-241	144	8.93	0.01	0.002	0.3	0.00001	8.64×10^{-9}	5.36×10^{-10}
Am-241	11.7	0.0694	0.01	0.002	0.3	0.00001	7.02×10^{-10}	4.16×10^{-12}
Am-242M	0.121	0.0000588	0.01	0.002	0.3	0.00001	7.26×10^{-12}	3.53×10^{-15}

[a] Radionuclide inventory from Appendix D.

[b] Use of data contained in the draft report (Goff et al. 1999a) for the driver spent nuclear fuel results in higher material-at-risk values for most isotopes presented in Table F–3 compared to data in the final report (Goff et al. 1999b). Therefore, these material-at- risk estimates were not revised to reflect data in the final report.

- *Cask drop and gaseous fission product release*—Spent nuclear fuel casks would be handled frequently when the sodium-bonded fuel is processed. (Spent nuclear fuel handling at the ANL-W site is not limited to that associated with the treatment of the sodium-bonded spent nuclear fuel. The accident discussed here is intended to address only that portion of the handling activity that can be directly attributed to the treatment of sodium-bonded spent nuclear fuel.) Spent nuclear fuel stored in the Radioactive Scrap and Waste Facility would be transferred to the Fuel Conditioning Facility for processing. Spent nuclear fuel would be received from off site at the Hot Fuel Examination Facility and would be transferred to the Fuel Conditioning Facility

for processing. The HFEF-5 cask would be used to move EBR-II driver and blanket spent nuclear fuel from the Radioactive Scrap and Waste Facility to the Fuel Conditioning Facility. The postulated accident is described in the Safety Analysis Report for the Hot Fuel Examination Facility (ANL 1998b). The accident involves a cask dropped during unloading, resulting in seal and fuel-cladding failure sufficient to release gaseous and volatile fission products to the atmosphere. The drop could be initiated by failure of lifting equipment, slings, hooks, cables, or human error by the lifting equipment operator. The cask drop was assumed conservatively to result in an unfiltered release of gaseous and volatile fission products. The release was assumed to be a puff release at ground level. Dropping of casks, while rare, is nevertheless categorized as anticipated, since such events have happened in the past and may be expected to occur over the lifetime of the facility. The frequency of cask dropping was assumed to be 0.01 per year, consistent with that used in the Safety Analysis Report for the Hot Fuel Examination Facility.

The HFEF-5 cask can contain two EBR-II driver spent nuclear fuel assemblies. It was assumed conservatively that the equivalent of one assembly (61 elements) fails in the accident. The material at risk, as shown in **Table F–4**, would be the equivalent of one EBR-II driver or blanket spent nuclear fuel assembly. The damage ratio for the failed elements was assumed to be 1, since all gaseous and volatile fission products conservatively could be released to the cask following cladding failure. The airborne release and respirable fractions for gases were each assumed to be 1, and 1×10^{-7} for cesium from the dislodgement of surface contamination (DOE 1995a). The accident was assumed to occur outdoors, with a leak path factor of 1.

Table F–4 Material at Risk and Release Fraction Values for a Cask Drop Accident at ANL-W

Material at Risk[a]							*Source Term (curies)*	
Isotope	*Blanket (curies)*	*Driver (curies)*	*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Blanket*	*Driver*
H-3 [b]	0.335	5.17	1	1	1	1	0.335	5.17
Kr-85	2.44	79.4	1	1	1	1	2.44	79.4
Cs-134	0.63	7.39	1	1.0×10^{-7}	1	1	6.30×10^{-8}	7.39×10^{-7}
Cs-137	81.3	928	1	1.0×10^{-7}	1	1	8.13×10^{-6}	0.0000928

[a] Data for one assembly based on Appendix D curie content data.
[b] It was assumed that 1 percent of this release becomes oxidized.

- *Salt transfer drop during movement from the Fuel Conditioning Facility to the Hot Fuel Examination Facility*—Solidified electrorefiner salt is sent from the Fuel Conditioning Facility to the Hot Fuel Examination Facility for processing into a final ceramic waste form. It is transferred in large chunks within the HFEF-5 cask. Transfer is via forklift or truck. In this scenario, a severe vehicle accident occurs, resulting in a breach of the inner and outer salt container. The accident could be caused by operator error or equipment failure. The accident is considered beyond-design-basis because of the durability of the shielded HFEF-5 canister. There would be over 200 transfers of salt from the Fuel Conditioning Facility to the Hot Fuel Examination Facility. A probability of 1×10^{-7} was assumed. The release occurs at ground level with a duration of one hour.

Table F–5 provides the isotopic material at risk for a total material at risk of 20 kilograms of salt. The salt is in chunks (i.e., ice cube-size) and is not combustible. No significant release was assumed from the large pieces. Some of the salt pieces would experience brittle fracture and release particulates. A brittle fracture particulate fraction for solidified salt would be 0.0001 for particles less than 10 microns in diameter (ANL 1998b); therefore, a damage ratio of 0.0001 was assumed. Conservatively, the same airborne release fraction and respirable fraction values were used for this scenario as for the salt powder spill in the Hot Fuel Examination Facility Main Cell; that is, the airborne release fraction for powder is 0.002 and the respirable fraction is 0.3 (DOE 1994b). The accident occurs outdoors; therefore, the leak path factor is 1.

Table F–5 Material at Risk and Release Fraction Values for a Salt Transfer Drop Accident at ANL-W

Material at Risk [a]			*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Source Term (curies)*	
Isotope	*Blanket (curies)*	*Driver (curies)*					*Blanket*	*Driver*
Sr-90	116	7,000	0.0001	0.002	0.3	1	6.96×10^{-6}	0.000420
Y-90	116	7,000	0.0001	0.002	0.3	1	6.96×10^{-6}	0.000420
I-129	0.000207	0.00261	0.0001	0.002	0.3	1	1.24×10^{-11}	1.57×10^{-10}
Cs-134	1.92	62.5	0.0001	0.002	0.3	1	1.15×10^{-7}	3.75×10^{-6}
Cs-137	249	7,850	0.0001	0.002	0.3	1	0.0000149	0.000471
Ba-137M	236	7,420	0.0001	0.002	0.3	1	0.0000142	0.000445
Ce-144	9.02	105	0.0001	0.002	0.3	1	5.41×10^{-7}	6.30×10^{-6}
Pr-144	9.02	105	0.0001	0.002	0.3	1	5.41×10^{-7}	6.30×10^{-6}
Pm-147	58.5	2,930	0.0001	0.002	0.3	1	3.51×10^{-6}	0.000176
Sm-151	14.4	190	0.0001	0.002	0.3	1	8.64×10^{-7}	0.0000114
Eu-154	1.06	20.1	0.0001	0.002	0.3	1	6.36×10^{-8}	1.21×10^{-6}
Eu-155	6.91	135	0.0001	0.002	0.3	1	4.15×10^{-7}	8.10×10^{-6}
Th-228	0.0000223	0.00183	0.0001	0.002	0.3	1	1.34×10^{-8}	1.10×10^{-10}
Np-237	0.00120	0.0103	0.0001	0.002	0.3	1	7.20×10^{-11}	6.18×10^{-10}
Pu-238	1.29	13.4	0.0001	0.002	0.3	1	7.74×10^{-8}	8.04×10^{-7}
Pu-239	103	21.6	0.0001	0.002	0.3	1	6.18×10^{-6}	1.30×10^{-6}
Pu-240	7.11	0.733	0.0001	0.002	0.3	1	4.27×10^{-7}	4.40×10^{-8}
Pu-241	28.8	1.79	0.0001	0.002	0.3	1	1.73×10^{-6}	1.07×10^{-7}
Am-241	2.34	0.0139	0.0001	0.002	0.3	1	1.40×10^{-7}	8.34×10^{-10}
Am-242M	0.0243	0.0000118	0.0001	0.002	0.3	1	1.46×10^{-9}	7.08×10^{-13}

[a] The material at risk is the isotope in 20 kilograms of salt, which is 20 percent of the values given in Table F-3.

- *Transuranic waste fire*—Transuranic waste is generated as a result of treatment operations, as well as other operations, at ANL-W. This waste is placed in containers and temporarily stored (staged) at ANL-W pending shipment to the Radioactive Waste Management Complex. A fire was postulated to occur in a 1.2 × 1.2 × 2.4-meter (4 × 4 × 8-foot) solid transuranic waste box because of spontaneous combustion, pyrophoric material, vehicle accident, electrical failure, or poor housekeeping. The fire consumes the contents of one box of transuranic waste. The accident was assumed to occur outdoors during handling. The release occurs at ground level over one hour. The Final Safety Analysis Report for the Fuel Conditioning Facility assigned an accident frequency in the range of 0.0001 to 0.01 (ANL 1998a). Here, the accident was assumed to have a frequency of 0.001 per year.

  The material at risk, as shown in **Table F–6**, was assumed to be one box of transuranic waste. The waste boxes are loaded with 1/20th of 0.34 curies of alpha activity, as described in the Fuel Conditioning Facility Final Safety Analysis Report (ANL 1998a). The material at risk is 0.017 curies of transuranic nuclides, with the nuclide distribution associated with the generic contents of a transuranic waste container. The damage ratio was assumed to be 1, since all waste in the container was assumed to be involved in the fire. An airborne release fraction of 0.0005 and a respirable fraction of 1 for burning of surface contaminated waste was used (DOE 1994b). The leak path factor was assumed to be 1. No credit was taken for building confinement.

Table F–6 Material at Risk and Release Fraction Values for a Transuranic Waste Fire Accident at ANL-W

Material at Risk [a]		*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Source Term (curies)*
Isotope	*Curies*					
Pu-238	0.000153	1	0.0005	1	1	7.67×10^{-8}
Pu-239	0.0123	1	0.0005	1	1	6.15×10^{-6}
Pu-240	0.000846	1	0.0005	1	1	4.23×10^{-7}
Pu-241	0.00343	1	0.0005	1	1	1.72×10^{-6}
Am-241	0.000266	1	0.0005	1	1	1.33×10^{-7}

[a] The material at risk is for a generic waste package, not for any specific spent nuclear fuel.

- *Design-basis earthquake - multifacility effects*—In the Fuel Conditioning Facility, the argon cell contains the equipment for processing sodium-bonded spent nuclear fuel into salt and metallic waste forms and a uranium metal product. All operations involving bare fuel are conducted in the argon cell because the inert atmosphere precludes pyrophoric metal fire. Fire cannot occur unless sufficient oxygen enters the cell through a cell breach. The walls, ceiling, and floor of the argon cell are constructed from reinforced concrete with thicknesses ranging from 1.2 to 1.5 meters (4 to 5 feet). It also has a gas-tight steel lining. It was assumed that the accident occurs during electrometallurgical treatment operations. Chopped fuel, electrorefiner salts, cathodes, and anodes are all present in the argon cell. Consistent with the assumption given in the Fuel Conditioning Facility Safety Analysis Report, a design-basis earthquake at this facility would result in a cell breach and in-leakage of air. The air in the cell would cause pyrophoric metals to ignite and burn. The Final Safety Analysis Report for the Fuel Conditioning Facility (ANL 1998a) identifies the seismic design goal for the facility to be the ability to withstand a 0.21 g design-basis earthquake. This event is identified as having a return frequency of 0.0002 per year. At this earthquake level, the electrorefiners are seismically qualified, and no spill of molten salt would occur. The safety exhaust system also would remain operational, although breaches could occur in the argon cell boundary after a design-basis earthquake. The safety exhaust building, which includes the high-efficiency particulate air filters, is designed to withstand an earthquake of 0.24 g, and was assumed to function as designed, filtering the cell atmosphere prior to release through the Fuel Conditioning Facility stack.

  In the Hot Fuel Examination Facility, grinding of salt into powder was assumed to be occurring in the Main Cell. The grinder is located in the Hot Fuel Examination Facility Main Cell on a raised floor consisting of steel plates resting on supports. Underneath the floor is a 2.4-meter-deep (8-foot-deep) pit that houses the ventilation ductwork and high-efficiency particulate air filters. At the Hot Fuel Examination Facility, a design-basis earthquake was assumed to cause the vessel containing ground salt to topple and the powder to spill out. Since the ventilation system was not seismically qualified, it was assumed to fail and result in an unfiltered release. It also was assumed that the design-basis earthquake would cause a loss of electrical power, which would result in failure of the ventilation system. The Main Cell breaches at piping or ventilation penetrations, providing a release path for the suspended powder. The releases occur over a one-hour period, and were modeled as a ground-level release.

  The Hot Fuel Examination Facility has been analyzed for a 0.14 g design-basis earthquake, an event with a return frequency of 0.001 per year and a performance goal of 0.0001 per year. The functionality of equipment after a 0.14 g earthquake has not been determined as yet. However, all major systems remained functional during the 0.03 g Borah earthquake in 1983, an event with a return frequency of 0.008 per year. While it is expected that the equipment would survive a 0.14 g earthquake with a frequency of 0.001 per year, the 0.008 per year earthquake frequency (ANL 1998b) was used conservatively to represent the upper bound of the design-basis earthquake, which would result in a salt powder spill and the failure of the ventilation system. This frequency is nearly two orders of magnitude higher than that corresponding to a 0.21 g earthquake that could impact both the Hot Fuel Examination Facility and the Fuel Conditioning

Facility. Therefore, 0.008 per year was used for the design-basis earthquake accident frequency for the Hot Fuel Examination Facility.

In the Fuel Conditioning Facility, the material at risk is chopped spent nuclear fuel and uranium in two electrorefiner cathodes in the argon cell at the time of the accident. **Table F–7** provides material-at-risk values for the isotopes of concern. The bounding inventory is 20 kilograms (44 pounds) of chopped fuel and the uranium in two solid electrorefiner cathodes (ANL 1998a). The solid cathodes contain 17 kilograms (37 pounds) of uranium. (Uranium is considered a toxic chemical in the chemical accident assessment, Section F.3.) The material at risk is, therefore, the 20 kilograms (44 pounds) of chopped spent nuclear fuel. For the metal fire in the argon cell, the damage ratio was assumed to be 1, since all materials in the material at risk are released to the cells in the accident. For the Fuel Conditioning Facility, the airborne release fraction values are 1 for krypton-85, 0.00025 for cesium, and 2.5×10^{-6} for strontium, uranium, and the transuranic waste nuclides; the respirable fractions are each 1 (DOE 1995a). For the Fuel Conditioning Facility, the safety exhaust system remains functional, and the release is filtered through high-efficiency particulate air filters. A leak path factor of 0.00001 was assumed for all particulates.

Table F–7 Material at Risk and Release Fraction Values for a Design-Basis Earthquake at ANL-W

Accident	*Material at Risk* [a]			*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Source Term (curies)*	
	Isotope	*Blanket (curies)*	*Driver (curies)*					*Blanket*	*Driver*
Design-basis earthquake and salt powder spill at the Hot Fuel Examination Facility	Sr-90	580	35,000	0.01	0.0020	0.30	0.125	0.000435	0.0263
	Y-90	580	35,000	0.01	0.0020	0.30	0.125	0.000435	0.0263
	I-129	0.00104	0.0131	0.01	0.0020	0.30	0.125	7.80×10^{-10}	9.83×10^{-9}
	Cs-134	9.63	313	0.01	0.0020	0.30	0.125	7.22×10^{-6}	0.000235
	Cs-137	1,240	39,200	0.01	0.0020	0.30	0.125	0.000930	0.0294
	Ba-137M	1,180	37,100	0.01	0.0020	0.30	0.125	0.000885	0.0278
	Ce-144	45.1	526	0.01	0.0020	0.30	0.125	0.0000338	0.000395
	Pr-144	45.1	526	0.01	0.0020	0.30	0.125	0.0000338	0.000395
	Pm-147	292	14,700	0.01	0.0020	0.30	0.125	0.000219	0.0110
	Sm-151	71.9	948	0.01	0.0020	0.30	0.125	0.0000539	0.000711
	Eu-154	5.28	101	0.01	0.0020	0.30	0.125	3.96×10^{-6}	0.0000758
	Eu-155	34.6	677	0.01	0.0020	0.30	0.125	0.0000260	0.000508
	Th-228	0.000111	0.00913	0.01	0.0020	0.30	0.125	8.33×10^{-11}	6.85×10^{-9}
	Np-237	0.00602	0.0513	0.01	0.0020	0.30	0.125	4.51×10^{-9}	3.85×10^{-8}
	Pu-238	6.44	66.8	0.01	0.0020	0.30	0.125	4.83×10^{-6}	0.0000501
	Pu-239	517	108	0.01	0.0020	0.30	0.125	0.000388	0.0000810
	Pu-240	35.5	3.67	0.01	0.0020	0.30	0.125	0.0000266	2.75×10^{-6}
	Pu-241	144	8.93	0.01	0.0020	0.30	0.125	0.000108	6.70×10^{-6}
	Am-241	11.7	0.0694	0.01	0.0020	0.30	0.125	8.78×10^{-6}	5.21×10^{-8}
	Am-242M	0.121	0.0000588	0.01	0.0020	0.30	0.125	9.08×10^{-8}	4.41×10^{-11}
Design-basis earthquake and metal fire in the Fuel Conditioning Facility argon cell	H-3	0.142	24.4	1	1	1	1	0.142	24.4
	C-14	0.00119	3,980	1	2.5×10^{-6}	1	0.00001	2.99×10^{-14}	9.95×10^{-8}
	Fe-55	1.80	97.4	1	2.5×10^{-6}	1	0.00001	4.51×10^{-11}	2.44×10^{-9}
	Co-60	0.318	9.62	1	2.5×10^{-6}	1	0.00001	7.95×10^{-12}	2.41×10^{-10}
	Ni-63	0.0612	4.58	1	2.5×10^{-6}	1	0.00001	1.53×10^{-12}	1.15×10^{-10}
	Kr-85	1.04	378	1	1	1	1	1.04	378
	Sr-90	16.1	3,940	1	2.5×10^{-6}	1	0.00001	4.04×10^{-10}	9.85×10^{-8}
	Y-90	16.1	3,940	1	2.5×10^{-6}	1	0.00001	4.04×10^{-10}	9.85×10^{-8}
	Ru-106	2.70	30.2	1	0.00025	1	0.00001	6.75×10^{-9}	7.55×10^{-8}
	Rh-106	2.70	30.2	1	2.5×10^{-6}	1	0.00001	6.75×10^{-11}	7.55×10^{-10}

Accident	Material at Risk [a]			Damage Ratio	Airborne Release Fraction	Respirable Fraction	Leak Path Factor	Source Term (curies)	
	Isotope	Blanket (curies)	Driver (curies)					Blanket	Driver
Design-basis earthquake and metal fire in the Fuel Conditioning Facility argon cell (cont'd)	Cd-113M	0.0142	0.928	1	2.5×10^{-6}	1	0.00001	3.56×10^{-13}	2.32×10^{-11}
	Sb-125	0.462	59.2	1	2.5×10^{-6}	1	0.00001	1.16×10^{-11}	1.48×10^{-9}
	Te-125M	0.190	24.6	1	2.5×10^{-6}	1	0.00001	4.76×10^{-12}	6.15×10^{-10}
	I-129	0.0000288	0.00147	1	1	1	1	0.0000288	0.00147
	Cs-134	0.268	35.2	1	0.00025	1	0.00001	6.70×10^{-10}	8.80×10^{-8}
	Cs-137	34.6	4,420	1	0.00025	1	0.00001	8.65×10^{-8}	0.0000111
	Ba-137M	32.8	4,180	1	2.5×10^{-6}	1	0.00001	8.20×10^{-10}	1.05×10^{-7}
	Ce-144	1.25	59.2	1	2.5×10^{-6}	1	0.00001	3.14×10^{-11}	1.48×10^{-9}
	Pr-144	1.25	59.2	1	2.5×10^{-6}	1	0.00001	3.14×10^{-11}	1.48×10^{-9}
	Pm-147	8.14	1,650	1	2.5×10^{-6}	1	0.00001	2.04×10^{-10}	4.13×10^{-8}
	Sm-151	2.00	107	1	2.5×10^{-6}	1	0.00001	5.00×10^{-11}	2.67×10^{-9}
	Eu-154	0.147	11.3	1	2.5×10^{-6}	1	0.00001	3.67×10^{-12}	2.84×10^{-10}
	Eu-155	0.962	76.2	1	2.5×10^{-6}	1	0.00001	2.41×10^{-11}	1.91×10^{-9}
	Th-228	3.10×10^{-6}	0.00103	1	2.5×10^{-6}	1	0.00001	7.75×10^{-17}	2.57×10^{-14}
	U-234	0.0000266	0.808	1	2.5×10^{-6}	1	0.00001	6.65×10^{-16}	2.02×10^{-11}
	U-235	0.0000754	0.0262	1	2.5×10^{-6}	1	0.00001	1.89×10^{-15}	6.55×10^{-13}
	U-236	0.0000848	0.0242	1	2.5×10^{-6}	1	0.00001	2.12×10^{-15}	6.05×10^{-13}
	U-238	0.00654	0.00222	1	2.5×10^{-6}	1	0.00001	1.64×10^{-13}	5.55×10^{-14}
	Np-237	2.60×10^{-6}	0.00578	1	2.5×10^{-6}	1	0.00001	6.50×10^{-17}	1.45×10^{-13}
	Pu-238	0.188	3.32	1	2.5×10^{-6}	1	0.00001	4.70×10^{-12}	8.30×10^{-11}
	Pu-239	15.1	5.38	1	2.5×10^{-6}	1	0.00001	3.77×10^{-10}	1.35×10^{-10}
	Pu-240	1.04	0.182	1	2.5×10^{-6}	1	0.00001	2.59×10^{-11}	4.56×10^{-12}
	Pu-241	4.20	0.444	1	2.5×10^{-6}	1	0.00001	1.05×10^{-10}	1.11×10^{-11}
	Am-241	0.326	0.00782	1	2.5×10^{-6}	1	0.00001	8.15×10^{-12}	1.96×10^{-13}
	Am-242M	0.00338	6.62×10^{-6}	1	2.5×10^{-6}	1	0.00001	8.45×10^{-14}	1.66×10^{-16}

[a] Radionuclide inventory from Appendix D.

During the postulated event, 100 kilograms (220 pounds) of solidified salt powder with the same concentration of radionuclides as described previously for the salt powder spill accident are assumed to be spilled in the Hot Fuel Examination Facility Main Cell. As previously discussed, in a salt powder spill, less than 1 percent of the salt would have the characteristics capable of resulting in an airborne release, i.e., a damage ratio of 0.01 was used. For the powder spill within the cell, an airborne release fraction of 0.002 and a respirable fraction of 0.3 were assumed (DOE 1994b). These are the same values as those used for the salt powder spill accident described previously. The Hot Fuel Examination Facility leak path for the release is through three enclosures before reaching the outside: the Main Cell, ducts and pipes, and the building. Consistent with the facility safety analysis report assumption, a leak path factor of 0.5 was assigned to each enclosure for plate-out and settling of the airborne powder. Therefore, the total leak path factor is $0.5 \times 0.5 \times 0.5 = 0.125$.

- *Aircraft crash*—The potential for an aircraft crash was evaluated. The methodology for evaluating the likelihood of an aircraft crash is documented in the *DOE Standard: Accident Analysis for Aircraft Crash into Hazardous Facilities* (DOE 1996c). At Idaho National Engineering and Environmental Laboratory (INEEL), the probabilities of a small and large aircraft crash are 2.3×10^{-4} and 1.0×10^{-6} crashes per square kilometer (9×10^{-5} and 4×10^{-7} crashes per square mile) per year, respectively. Using guidance in this DOE standard, the effective area of the Fuel Conditioning Facility was calculated accounting for aircraft wing span and potential skid distance. The effective area of the Fuel Conditioning Facility is about 0.078 square kilometers (0.03 square miles) for a large aircraft, and 0.018 square kilometers (0.007 square miles) for a

small aircraft. The effective area of the Fuel Conditioning Facility is conservative because the combined area of the air and argon cells, where the hazardous materials are contained, is smaller than the total area of the building. Multiplying the effective area by INEEL-specific crash rates gives an estimated probability of a crash into the Fuel Conditioning Facility of 1×10^{-8} for large aircraft and 6×10^{-7} for small aircraft. Comparable probabilities are applicable to the Hot Fuel Examination Facility. A large aircraft crash is not reasonably foreseeable, and given the 1.2 to 1.5-meter-thick (4 to 5-foot-thick) walls of the cells and the "buffer" provided by the building exterior walls, the crash of a small aircraft is unlikely to result in any damage to the cells. Damage from the more probable seismic events analyzed is considered to bound the damage that could result from a small aircraft crash. Also, seismic events affect more than one facility, while an aircraft crash could affect only one facility. Therefore, an aircraft crash was not analyzed separately.

- *Nuclear criticality*—The potential for a nuclear criticality was considered in the accident analysis. Nuclear criticality has been evaluated in the safety analyses documented for the ANL-W facilities, as required by DOE. The existing safety analyses conclude that nuclear criticality is beyond the design-basis of the facilities proposed for the electrometallurgical treatment alternative and, therefore, has a probability of less than 1×10^{-6} per year. This conclusion is based on a lack of nuclear moderator materials, equipment design, and inventory controls, as well as numerous other administrative controls and operating procedures. The intent of the process is to dilute, rather than concentrate, fissile materials. Fuel storage racks and processing equipment are designed to maintain their safety function during the design-basis earthquake. Even in a beyond-design-basis earthquake (maximum frequency of 0.00001 per year), nuclear materials would have to come together in an ideal critical array for criticality to be possible. For example, it would require more than the equivalent of 10 EBR-II driver spent nuclear fuel assemblies (610 individual elements) in an ideal geometric configuration to create a potential criticality hazard. During processing, some fuel would be stored in the hot cells. This fuel is stored in the storage cans within the floor storage pits. The floor storage pits are evenly spaced 61 centimeters (2 feet) from the center, located almost entirely on a 3-meter-thick (10-foot-thick) hot cell concrete floor. These pits are designed to maintain criticality-safe configurations under all normal and design-basis abnormal conditions, including a design-basis earthquake (ANL 1998a). An evaluation of earthquake loading has concluded that no uplifting of the hot cell floor would occur in a beyond-design-basis earthquake of 0.3 g peak ground acceleration (corresponding to an earthquake frequency of 0.00001 per year) (ANL 1999). Therefore, the conditional probability of creating a criticality hazard configuration, given a beyond-design-basis earthquake, was estimated to be no greater than 0.01. Therefore, criticality is not considered to be reasonably foreseeable, and was not analyzed quantitatively.

- *Beyond-design-basis earthquake*—This scenario is similar to the design-basis earthquake except that the safety exhaust system was not assumed to function at the Fuel Conditioning Facility, and an electrorefiner was assumed to spill its molten salt. Also, since spent nuclear fuel elements are stored in both the Fuel Conditioning Facility and the Hot Fuel Examination Facility, a fraction of stored fuel elements were assumed to experience cladding failure and release of gaseous and volatile fission products. All releases were modeled as ground-level releases. The Fuel Conditioning Facility horizontal acceleration design-basis is 0.21 g, and the newer safety equipment building is designed for a 0.24 g horizontal acceleration. A 0.24 g peak acceleration corresponds to an earthquake frequency at ANL-W of approximately 0.0001 per year (WCFS 1998). The Fuel Conditioning Facility natural phenomena hazard performance goal is a frequency of 0.00001 (DOE 1994a). (The Hot Fuel Examination Facility performance goal is 0.0001.) The performance goal can be interpreted as the frequency level at which facility damage will initiate. The Fuel Conditioning Facility and safety exhaust system are not expected to suffer damage from earthquakes with frequencies higher than this. Therefore, the upper bound for the beyond-design-basis earthquake frequency was assumed to correspond to the frequency of the performance goal, 0.00001per year.

  The material at risk, provided in **Table F–8**, would be the same as for the design-basis earthquake, with the addition of the salt in the electrorefiners and the fuel elements and subassemblies in storage. Although the electrorefiners are seismically qualified, one of the two electrorefiners in the Fuel Conditioning Facility argon cell was assumed conservatively to spill its molten salt. It was assumed that approximately 700 kilograms (1,540 pounds) of salt are fully loaded with radionuclides from the processing of 5.56 metric tons of heavy metal of blanket spent nuclear fuel elements or 1.1 metric tons of driver spent nuclear fuel

elements and are about to be replaced at the time of the accident. The damage ratio for all but the fuel assemblies in storage was assumed to be 1, as in the design-basis earthquake. Both the blanket and driver spent nuclear fuel elements are stored in racks with the cladding intact. In the earthquake, some could be expected to fall out of the racks or be hit by falling debris, but it is not reasonable to assume all assemblies would be damaged. It was assumed that 10 percent of the elements stored in the cells at the time of the earthquake experience cladding failure and release gaseous and volatile fission products. For the driver spent nuclear fuel elements, this is the equivalent of 12 driver assemblies (or 50 kilograms [110 pounds] of heavy metal). Ten percent of the stored blanket elements is the equivalent of 370 kilograms (771 pounds) of heavy metal. The airborne release fraction and respirable fraction values are the same as for the design-basis earthquake, with the addition of krypton and cesium from the failed EBR-II driver and blanket fuel. The airborne release fraction and respirable fraction values for krypton and tritium (H-3), both elements in the gaseous state, are each 1. For the molten salt spill, the airborne release fraction and respirable fraction values for viscous solutions (DOE 1994b) were used: 4×10^{-6} (0.0004 for iodine and cesium) for the airborne release fraction and 0.8 for the respirable fraction. The forces associated with the beyond-design-basis earthquake were assumed to result in the failure of confinement integrity. The cells were assumed to experience major failure, and the release would be directly to the atmosphere. The leak path factor is 1.

Table F–8 Material at Risk and Release Fraction Values for a Beyond-Design-Basis Earthquake at ANL-W

Accident	*Material at Risk*[a]			*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Source Term (curies)*	
	Isotope	*Blanket (curies)*	*Driver (curies)*					*Blanket*	*Driver*
Beyond-design-basis earthquake and salt powder spill in the Hot Fuel Examination Facility	Sr-90	580	35,000	0.01	0.002	0.3	1	0.00348	0.21
	Y-90	580	35,000	0.01	0.002	0.3	1	0.00348	0.21
	I-129	0.00104	0.0131	0.01	0.002	0.3	1	6.24×10^{-9}	7.86×10^{-8}
	Cs-134	9.63	313	0.01	0.002	0.3	1	0.0000578	0.00188
	Cs-137	1,240	39,200	0.01	0.002	0.3	1	0.00744	0.235
	Ba-137M	1,180	37,100	0.01	0.002	0.3	1	0.00708	0.223
	Ce-144	45.1	526	0.01	0.002	0.3	1	0.000271	0.00316
	Pr-144	45.1	526	0.01	0.002	0.3	1	0.000271	0.00316
	Pm-147	292	14,700	0.01	0.002	0.3	1	0.00175	0.0882
	Sm-151	71.9	948	0.01	0.002	0.3	1	0.000431	0.00569
	Eu-154	5.28	101	0.01	0.002	0.3	1	0.0000317	0.000606
	Eu-155	34.6	677	0.01	0.002	0.3	1	0.000148	0.00406
	Th-228	0.000111	0.00913	0.01	0.002	0.3	1	6.66×10^{-10}	5.48×10^{-8}
	Np-237	0.00602	0.0513	0.01	0.002	0.3	1	3.61×10^{-8}	3.08×10^{-7}
	Pu-238	6.44	66.8	0.01	0.002	0.3	1	0.0000386	0.000401
	Pu-239	517	108	0.01	0.002	0.3	1	0.00310	0.000648
	Pu-240	35.5	3.67	0.01	0.002	0.3	1	0.000213	0.000022
	Pu-241	144	8.93	0.01	0.002	0.3	1	0.000864	0.0000536
	Am-241	11.7	0.0694	0.01	0.002	0.3	1	0.0000702	4.16×10^{-7}
	Am-242M	0.121	0.0000588	0.01	0.002	0.3	1	7.26×10^{-7}	3.53×10^{-10}
Beyond-design-basis earthquake and metal fire in the Fuel Conditioning Facility argon cell	H-3	0.142	24.4	1	1	1	1	0.142	24.4
	C-14	0.00119	3,980	1	2.5×10^{-6}	1	1	2.99×10^{-9}	0.00995
	Fe-55	1.80	97.4	1	2.5×10^{-6}	1	1	4.51×10^{-6}	0.000244
	Co-60	0.318	9.62	1	2.5×10^{-6}	1	1	7.95×10^{-7}	0.0000241
	Ni-63	0.0612	458	1	2.5×10^{-6}	1	1	1.53×10^{-7}	0.00115
	Kr-85	1.04	378	1	1	1	1	1.04	378
	Sr-90	16.1	3,940	1	2.5×10^{-6}	1	1	0.0000404	0.00985
	Y-90	16.1	3,940	1	2.5×10^{-6}	1	1	0.0000404	0.00985
	Ru-106	2.70	30.2	1	0.00025	1	1	0.000675	0.00755

Accident	Material at Risk[a]			Damage Ratio	Airborne Release Fraction	Respirable Fraction	Leak Path Factor	Source Term (curies)	
	Isotope	Blanket (curies)	Driver (curies)					Blanket	Driver
Beyond-design-basis earthquake and metal fire in the Fuel Conditioning Facility argon cell (cont'd)	Rh-106	2.70	30.2	1	2.5×10^{-6}	1	1	6.75×10^{-6}	0.0000755
	Cd-113M	0.0142	0.928	1	2.5×10^{-6}	1	1	3.56×10^{-8}	2.32×10^{-6}
	Sb-125	0.462	59.2	1	2.5×10^{-6}	1	1	1.15×10^{-6}	0.000148
	Te-125M	0.190	24.6	1	2.5×10^{-6}	1	1	4.76×10^{-7}	0.0000615
	I-129	0.0000288	0.00147	1	1	1	1	0.0000288	0.00147
	Cs-134	0.268	35.2	1	0.00025	1	1	0.000067	0.00880
	Cs-137	34.6	4,420	1	0.00025	1	1	0.00865	1.10
	Ba-137M	32.8	4,180	1	2.5×10^{-6}	1	1	0.000082	0.0105
	Ce-144	1.25	59.2	1	2.5×10^{-6}	1	1	3.14×10^{-6}	0.000148
	Pr-144	1.25	59.2	1	2.5×10^{-6}	1	1	3.14×10^{-6}	0.000148
	Pm-147	8.14	1,650	1	2.5×10^{-6}	1	1	0.0000204	0.00413
	Sm-151	2.00	107	1	2.5×10^{-6}	1	1	5.00×10^{-6}	0.000267
	Eu-154	0.147	11.3	1	2.5×10^{-6}	1	1	3.67×10^{-7}	0.0000284
	Eu-155	0.962	76.2	1	2.5×10^{-6}	1	1	2.41×10^{-6}	0.000191
	Th-228	3.10×10^{-6}	0.00103	1	2.5×10^{-6}	1	1	7.75×10^{-12}	2.57×10^{-9}
	Np-237	0.0000266	0.808	1	2.5×10^{-6}	1	1	6.65×10^{-11}	2.02×10^{-6}
	U-234	0.0000754	0.0262	1	2.5×10^{-6}	1	1	1.89×10^{-10}	6.55×10^{-8}
	U-235	0.0000848	0.0242	1	2.5×10^{-6}	1	1	2.12×10^{-10}	6.05×10^{-8}
	U-236	0.00654	0.00222	1	2.5×10^{-6}	1	1	1.64×10^{-8}	5.55×10^{-9}
	U-238	2.60×10^{-6}	0.00578	1	2.5×10^{-6}	1	1	6.50×10^{-12}	1.45×10^{-8}
	Pu-238	0.188	3.32	1	2.5×10^{-6}	1	1	4.70×10^{-7}	8.30×10^{-6}
	Pu-239	15.1	5.38	1	2.5×10^{-6}	1	1	0.0000377	0.0000135
	Pu-240	1.04	0.182	1	2.5×10^{-6}	1	1	2.59×10^{-6}	4.56×10^{-7}
	Pu-241	4.20	0.444	1	2.5×10^{-6}	1	1	0.0000105	1.11×10^{-6}
	Am-241	0.326	0.00782	1	2.5×10^{-6}	1	1	8.15×10^{-7}	1.96×10^{-8}
	Am-242M	0.00338	6.62×10^{-6}	1	2.5×10^{-6}	1	1	8.45×10^{-9}	1.66×10^{-11}
Beyond-design-basis earthquake and liquid salt spill at the Fuel Conditioning Facility	Sr-90	4,490	245,000	1	4.0×10^{-6}	0.8	1	0.0144	0.784
	Y-90	4,490	245,000	1	4.0×10^{-6}	0.8	1	0.0144	0.784
	I-129	0.00801	0.0917	1	0.0004	0.8	1	2.56×10^{-6}	0.0000293
	Cs-134	74.5	2,190	1	0.0004	0.8	1	0.0238	0.701
	Cs-137	9,620	274,000	1	0.0004	0.8	1	3.08	87.8
	Ba-137M	9,120	260,000	1	4.0×10^{-6}	0.8	1	0.0292	0.831
	Ce-144	349	3,680	1	4.0×10^{-6}	0.8	1	0.00112	0.0118
	Pr-144	349	3,680	1	4.0×10^{-6}	0.8	1	0.00112	0.0118
	Pm-147	2,260	103,000	1	4.0×10^{-6}	0.8	1	0.00723	0.329
	Sm-151	556	6,640	1	4.0×10^{-6}	0.8	1	0.00178	0.0212
	Eu-154	40.8	707	1	4.0×10^{-6}	0.8	1	0.000131	0.00226
	Eu-155	267	4,740	1	4.0×10^{-6}	0.8	1	0.000854	0.0152
	Th-228	0.000862	0.0639	1	4.0×10^{-6}	0.8	1	2.76×10^{-9}	2.05×10^{-7}
	Np-237	0.0465	0.359	1	4.0×10^{-6}	0.8	1	1.49×10^{-7}	1.15×10^{-6}
	Pu-238	49.8	468	1	4.0×10^{-6}	0.8	1	0.000159	0.0015
	Pu-239	4,000	756	1	4.0×10^{-6}	0.8	1	0.0128	0.00242
	Pu-240	274	25.7	1	4.0×10^{-6}	0.8	1	0.000877	0.0000822
	Pu-241	1,110	62.5	1	4.0×10^{-6}	0.8	1	0.00355	0.0002
	Am-241	90.6	0.486	1	4.0×10^{-6}	0.8	1	0.000290	1.55×10^{-6}
	Am-242M	0.940	0.000412	1	4.0×10^{-6}	0.8	1	3.01×10^{-6}	1.32×10^{-9}

Accident	Material at Risk[a]			Damage Ratio	Airborne Release Fraction	Respirable Fraction	Leak Path Factor	Source Term (curies)	
	Isotope	Blanket (curies)	Driver (curies)					Blanket	Driver
Beyond-design-basis earthquake and stored fuel assembly cladding failure	H-3	2.64	62	1	1	1	1	2.64	62
	Kr-85	19.1	953	1	1	1	1	19.1	953

[a] Radionuclide Inventory from Appendix D.

Accident Scenario Descriptions for Melt and Dilute Processing—The melt and dilute process would occur in the Hot Fuel Examination Facility hot cell at ANL-W. Two melt and dilute process options are considered for ANL-W: (1) cleaning blanket spent nuclear fuel (removing metallic sodium), and (2) cleaning blanket and driver (to the extent possible) spent nuclear fuel (see Appendix C for more details). Sufficient steel would be added to both process options to form an alloy with a composition of 50 percent each of uranium and steel. Both options would occur at a temperature range of about 1,400 °C (2,550 °F). For analysis purposes, it was assumed that, on average, 120 batches of melt and dilute processing could be performed per year, considering an 80 percent availability and a three-batches-per-week operation. Each batch would process about 60 kilograms (132 pounds) of heavy metal of blanket spent nuclear fuel or about 16 kilograms (35 pounds) of driver spent nuclear fuel (diluted with depleted uranium to a 60-kilogram-equivalent [132-pound-equivalent] heavy metal). This would lead to eight years of operations for processing blanket spent nuclear fuel and two years of processing for driver spent nuclear fuel. Prior to the melt and dilute process, the sodium-bonded spent fuel elements would be cut into segments. The segmented fuel elements would be heated to a temperature above the 200 °C (392 °F) melting point of sodium and the molten sodium would be drained into a collection tank. The temperature of this bulk sodium would be raised to 690 °C (1,274 °F), to volatilize the cesium and separate it from the sodium (see Appendix C for a more detailed description of this process).

Table F–9 identifies a list of accident scenarios that were considered to be applicable to the melt and dilute process as proposed at ANL-W. These scenarios are based on the analysis of the melt and dilute process provided in the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000). The accident scenarios and the corresponding source terms have been modified to reflect the specifics associated with the design of the Hot Fuel Examination Facility, the characteristics of the fuel type being processed, the material at risk, and the related release fractions.

Table F–9 Selected Accident Scenarios for Melt and Dilute Processing at ANL-W

Scenario	Frequency (per year)
Nuclear criticality	0.0003
Cask drop	0.01
Waste handling accident	0.0024
Sodium fire [a]	0.008
Aircraft crash	6×10^{-7} to 1×10^{-8}
Design-basis earthquake	0.008

[a] This event is evaluated as being a direct consequence of the design-basis earthquake.

Each accident scenario description sets the condition of the accident and provides a summary of the material involved. The following paragraphs provide a summary of the accidents analyzed, the material at risk, and the release factors for the EBR-II blanket and driver spent nuclear fuel (the Fermi-1 blanket spent nuclear fuel has a very low radioactive inventory).

- *Nuclear criticality*—A criticality accident could result from the processing of multiple batches (double batching) of fissile material in the melter. This accident was considered for the driver spent nuclear fuel only. The criticality was assumed to consist of 5×10^{17} fissions (a solid criticality fission yield) (DOE 2000). The Hot Fuel Examination Facility structure would not be compromised and its ventilation system would be expected to continue to function after a criticality event. Procedural controls would be used to prevent such an accident. Therefore, such an accident would be the result of a combination of human errors, as all criticality controls are designed to meet double contingency requirements. The Hot Fuel Examination Facility Safety Analysis Report identifies a criticality event as an incredible event with an assigned frequency of less than 1×10^{-6} per year (ANL 1998b). However, this Safety Analysis Report does not specifically address melt and dilute operations. A criticality event for the SRS melt and dilute process has been addressed (DOE 2000) and, for consistency among alternatives, this analysis has been adapted. Based on the assumption of approximately 120 batches of melt and dilute operations per year and a similar frequency analysis for this type of accident at SRS, the expected frequency of this event was estimated to be 0.0003 per year for the melt and dilute operations at ANL-W. The material at risk and release fractions are provided in **Table F–10**. The damage ratio and leak path factor for the volatile, gaseous fission products were assumed conservatively to be 1. A respirable fraction value of 1 also was used. The airborne release fraction values range from 0.5 to 0.05 (DOE 1994b).

Table F–10 Melt and Dilute Process Material at Risk and Release Fraction Values for a Nuclear Criticality Event at ANL-W

Material at Risk		*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Source Term (curies)*
Isotope	*Curies*					
Br-83	4.90	1	0.05	1	1	0.25
Br-84	16.3	1	0.05	1	1	0.82
Kr-83M	1.50	1	0.5	1	1	0.75
Kr-85M	7.2	1	0.5	1	1	3.6
Kr-87	32.8	1	0.5	1	1	17
Kr-88	32.9	1	0.5	1	1	17
Kr-89	1820	1	0.5	1	1	910
Te-129	2.70	1	0.07	1	1	0.19
Te-131	57.5	1	0.07	1	1	4.0
Te-131M	0.320	1	0.07	1	1	0.022
Te-132	1.60	1	0.07	1	1	0.11
Te-133	25.7	1	0.07	1	1	1.8
Te-133M	30.3	1	0.07	1	1	2.1
Te-134	90.5	1	0.07	1	1	6.3
I-131	0.212	1	0.05	1	1	0.011
I-132	0.855	1	0.05	1	1	0.043
I-133	6.80	1	0.05	1	1	0.34
I-134	98.0	1	0.05	1	1	4.9
I-135	22.1	1	0.05	1	1	1.1
Xe-133	0.026	1	0.5	1	1	0.013
Xe-135	2.61	1	0.5	1	1	1.3
Xe-135M	23.9	1	0.5	1	1	0.12
Xe-137	1940	1	0.5	1	1	0.097
Xe-138	665	1	0.5	1	1	0.033

- *Cask drop*—This event is similar to the cask drop event analyzed for the electrometallurgical treatment process. Spent nuclear fuel casks would be handled frequently when the sodium-bonded fuel is treated using the melt and dilute process. (Spent nuclear fuel handling at the ANL-W site is not limited to that associated

with the treatment of the sodium-bonded spent nuclear fuel. The accident discussed here is intended to address only that portion of the handling activity that can be directly attributed to the treatment of sodium-bonded spent nuclear fuel.) The accident involves a dropped cask during loading or unloading, seal failure, and spent nuclear fuel cladding failure sufficient to release gaseous and volatile fission products to the atmosphere, and is the same as previously described for the cask drop accident for the electrometallurgical treatment process. The material at risk and release fraction values are provided in Table F–4. (See the accident description for more detail.)

- *Waste handling accident*—The filters used in the melt and dilute off-gas exhaust system must be cleaned periodically and the resultant liquid waste disposed of. Decontamination of the filters was assumed to be performed after 10 batches are processed. Therefore, it was assumed that after processing 600 kilograms (1,320 pounds) of heavy metal of blanket spent nuclear fuel or 160 kilograms (352 pounds) of heavy metal of driver spent nuclear fuel, the filters would be decontaminated. It was postulated that a spill would occur during the transfer of the decontaminated liquid from one container to another. The event frequency is estimated at 0.0024 events per year (WSRC 1998a). The material at risk is from the fission products released during the melting process and collected on the filters. This includes fission products with boiling points at or below 1400 °C (2,550 °F) and some metal oxides that can be expected to form during the heating process (WSRC 1998b). A damage ratio of 0.5 was assumed to account for the spilling of half of the material during the accident. Airborne release fraction and respirable fraction values of 0.0002 and 0.5, respectively, were chosen for the material based on the release of material from aqueous spills (DOE 1994b). The spill was assumed to occur in an area not provided with a filtration system and, therefore, the leak path factor is 1. The material at risk, release fractions, and curies released for this accident for both EBR-II blanket and driver spent nuclear fuel are presented in **Table F–11**.

Table F–11 Melt and Dilute Process Material at Risk and Release Fraction Values for a Waste Handling Accident at ANL-W

Material At Risk [a]			*Damage Ratio* [b]	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Source Term (curies)*	
Isotope	*Blanket (curies)*	*Driver (curies)*					*Blanket*	*Driver*
Sr-90	484.2	31520	0.5	0.0002	0.5	1	0.024	1.58
Sb-125	13.86	473.6	0.5	0.0002	0.5	1	0.00069	0.024
Te-125M	5.71	196.8	0.5	0.0002	0.5	1	0.00029	0.0098
I-129	0.00086	0.012	0.5	0.0002	0.5	1	4.32×10^{-8}	6.0×10^{-7}
Cs-134	8.04	281.6	0.5	0.0002	0.5	1	0.000402	0.014
Cs-137	1038	35360	0.5	0.0002	0.5	1	0.0519	1.77
Pu-238	5.63	26.6	0.000015	0.0002	0.5	1	8.4×10^{-9}	4.0×10^{-8}
Pu-239	451.8	43.0	0.000015	0.0002	0.5	1	6.8×10^{-7}	6.5×10^{-8}
Pu-240	31.08	1.5	0.000015	0.0002	0.5	1	4.7×10^{-8}	2.3×10^{-9}
Pu-241	126.0	3.6	0.000015	0.0002	0.5	1	1.9×10^{-7}	5.4×10^{-9}
Am-241	9.78	0.063	0.000015	0.0002	0.5	1	1.5×10^{-8}	9.5×10^{-11}
Am-242M	0.10	0.000016	0.000015	0.0002	0.5	1	1.5×10^{-10}	2.4×10^{-14}

[a] Radionuclide inventory from Appendix D.
[b] Damage ratio values for particulates that would not be condensed in the off-gas system include a factor of 0.00003 to account for the fraction oxidized and released from liquid metals and captured on the filters.

- *Aircraft crash*—The potential for an aircraft crash was evaluated for the Hot Fuel Examination Facility and the Fuel Conditioning Facility as part of the evaluation of the electrometallurgical treatment process. The discussion provided previously is applicable to the use of the Hot Fuel Examination Facility in the melt and dilute process (see the discussion for the electrometallurgical treatment process earlier in this section). It was

concluded that the likelihood of an aircraft crash causing damage to the facility process is not reasonably foreseeable; therefore, no specific analysis is needed.

- *Sodium Fire*—The sodium fire event selected for analysis was postulated to occur during the fuel cleaning (sodium removal) process for the sodium-bonded spent nuclear fuel. The event is the result of a breach in the Hot Fuel Examination Facility cell followed by a sodium fire. This event can occur as a result of the design-basis earthquake, which results in Main Cell breaches at piping and ventilation penetrations and results in a failure of the ventilation system. The frequency of this event is 0.008 per year (or once in 125 years).

  It has been estimated that approximately 10 percent of the cesium in the spent nuclear fuel has migrated from the fuel region and bonded with the sodium being removed in the fuel cleaning process. Using the radionuclide inventories provided in Appendix D for the EBR-II driver and radial blanket spent nuclear fuel and the Fermi-1 blanket spent nuclear fuel, it was estimated that a total of 670 curies of cesium-134 and 76,000 curies of cesium-137 would be entrained within the sodium. Assuming that as much as one-half of the sodium is accumulated within the collection tank prior to processing to remove cesium from the sodium, the material at risk for the sodium fire would be 340 curies of cesium-134 and 38,000 curies of cesium-137. The release fractions selected for this accident are a damage ratio of 1, an airborne release fraction and a respirable fraction each of 0.00025, and a leak path factor of 0.125. The airborne release fraction and respirable fraction value is the same as that used for cesium release from a metal fire in the design-basis seismic event analysis. The leak path factor is the value used for the Hot Fuel Examination Facility during a design-basis seismic event. The total quantity of cesium released (source term) as a result of this accident is 0.011 curies of cesium-134 and 1.2 curies of cesium-137. The cesium source term from sodium in driver fuel is 0.0095 curies of cesium-134 and 1.14 curies of cesium-137.

- *Design-basis earthquake*—This is the same accident that was developed for the design-basis earthquake for the electrometallurgical treatment process at the Hot Fuel Examination Facility. The equipment availability and damage were assumed to be the same when the facility is used for the melt and dilute process as when it is used for the electrometallurgical treatment process. Consistent with the facility safety analysis report, the ventilation system was assumed to have failed, creating a leak path factor of 0.125. The frequency of this event is 0.008 per year (or once in 125 years).

  The damage ratio, airborne release fraction, respirable fraction, and leak path factor are the same as for the electrometallurgical treatment process design-basis earthquake, with a few exceptions. Because the melt and dilute process at ANL-W operates at an elevated temperature of about 1,400 °C (2,550 °F), some fission products would boil off during the process and enter the off-gas control system. The airborne release fraction for these volatilized fission product materials, e.g., strontium, antimony, cesium, tellurium, and iodine, is set at 1 (DOE 1994b). In addition, even though some of these materials could have been condensed and removed from the off-gas system at the time of the accident, it was assumed conservatively that all of these materials would be volatilized upon initiation of the accident. However, credit is taken for the condensation of these gases as they pass through the structures on the release path. These gases will cool from their initial release temperatures as they pass through the relatively cooler structures of the Hot Fuel Examination Facility. A factor of 0.5 was used for each structure, resulting in an airborne release fraction value (representing the fraction released to the atmosphere from the cell atmosphere) of 0.125. Gaseous krypton and tritium (H-3) were not considered here, since they were assumed to have been released to the environment during the fuel cleaning process. The source terms and release fractions are provided in **Table F–12**.

Table F–12 Melt and Dilute Process Material at Risk and Release Fraction Values for a Design-Basis Earthquake at ANL-W

Material At Risk *			Damage Ratio	Airborne Release Fraction	Respirable Fraction	Leak Path Factor	Source Term (curies)	
Isotope	Blanket (curies)	Driver (curies)					Blanket	Driver
Co-60	0.95	7.70	1	4.0×10^{-6}	0.8	0.125	3.8×10^{-7}	3.1×10^{-6}
Sr-90	48.4	3152	1	0.125	1	0.125	0.76	49
Y-90	48.4	3152	1	4.0×10^{-6}	0.8	0.125	0.000019	0.0013
Ru-106	8.1	24.16	1	4.0×10^{-6}	0.8	0.125	3.2×10^{-6}	9.8×10^{-6}
Rh-106	8.1	24.16	1	4.0×10^{-6}	0.8	0.125	3.2×10^{-6}	9.8×10^{-6}
Cd-113M	0.043	0.74	1	4.0×10^{-6}	0.8	0.125	1.7×10^{-8}	3.0×10^{-7}
Sb-125	1.39	47.36	1	0.125	1	0.125	0.022	0.74
Te-125M	0.57	19.68	1	0.125	1	0.125	0.0089	0.31
I-129	0.000086	0.0012	1	0.125	1	0.125	1.3×10^{-6}	0.000019
Cs-134	0.80	28.16	1	0.125	1	0.125	0.013	0.44
Cs-137	103.8	3536.0	1	0.125	1	0.125	1.6	55
Ba-137M	98.4	3344.0	1	4.0×10^{-6}	0.8	0.125	0.000039	0.0013
Ce-144	3.76	47.36	1	4.0×10^{-6}	0.8	0.125	1.5×10^{-6}	0.000019
Pr-144	3.76	47.36	1	4.0×10^{-6}	0.8	0.125	1.5×10^{-6}	0.000019
Pm-147	24.4	1321.6	1	4.0×10^{-6}	0.8	0.125	9.8×10^{-6}	0.00053
Sm-151	6.0	85.44	1	4.0×10^{-6}	0.8	0.125	2.4×10^{-6}	0.000034
Eu-154	0.44	9.07	1	4.0×10^{-6}	0.8	0.125	1.8×10^{-7}	3.6×10^{-6}
Eu-155	2.89	60.96	1	4.0×10^{-6}	0.8	0.125	1.2×10^{-6}	0.000024
Pu-238	0.56	2.66	1	4.0×10^{-6}	0.8	0.125	2.2×10^{-7}	1.1×10^{-6}
Pu-239	45.18	4.30	1	4.0×10^{-6}	0.8	0.125	0.000018	1.7×10^{-6}
Pu-240	3.11	0.15	1	4.0×10^{-6}	0.8	0.125	1.2×10^{-6}	6.0×10^{-8}
Pu-241	12.6	0.36	1	4.0×10^{-6}	0.8	0.125	5.0×10^{-6}	1.4×10^{-7}
Am-241	0.98	0.0063	1	4.0×10^{-6}	0.8	0.125	3.9×10^{-7}	2.5×10^{-9}
Am-242M	0.010	1.6×10^{-6}	1	4.0×10^{-6}	0.8	0.125	4.0×10^{-9}	6.4×10^{-13}

* The material at risk is the content in one batch: 60 kilograms of heavy metal of blanket fuel or 16 kilograms of heavy metal of driver fuel. Radionuclide inventory from Appendix D.

F.2.2.1.3 Accident Scenario Descriptions and Source Terms at SRS

Accident Scenario Descriptions for Plutonium-Uranium Extraction (PUREX) Processing—The following facilities would be used to store or process sodium-bonded spent nuclear fuel at SRS: F-Canyon, FB-Line, and the plutonium storage facility. The F-Canyon, FB-Line, and plutonium storage facility are part of the Building 221-F (or F-Canyon) structure. Shipments of the declad and cleaned blanket spent nuclear fuel cannot be received directly at the F-Canyon facility. The facility is not equipped to handle the transportation casks being used. The shipments would be received at the L-Reactor disassembly basin, transferred to casks suitable for shipment to F-Canyon, and then moved to F-Canyon. The PUREX process can be used to separate the plutonium from the blanket spent nuclear fuel pins. In the PUREX process, the declad and cleaned blanket spent nuclear fuel would be dissolved in the F-Canyon dissolvers, and fission products would be separated from uranium and plutonium. The plutonium solution then would be pumped to the FB-Line for purification and solidification. The depleted uranium solution would be pumped to A-Line tanks for storage and future processing into depleted uranium oxides.

The accident scenarios, identified in **Table F–13** and defined in the following paragraphs, are applicable to the processing facilities as a whole (i.e., F-Canyon and FB-Line). Transfer and storage accidents also were

considered for the analysis of F-Canyon-related activities. The sodium-bonded spent nuclear fuel would be declad and cleaned prior to shipment from ANL-W. This process results in the release of gases in the gap between the fuel and cladding (see Appendix E, Section E.4), the dominant radionuclides considered during the analysis of transfer (fuel and cask drop) accidents. Therefore, the accidents were not quantified. Accidents associated with storage of the sodium-bonded spent nuclear fuel and storage of the process products (plutonium and various waste forms) were assessed as having no additional impacts beyond those associated with the process-related accidents.

Table F–13 Selected Accident Scenarios for PUREX Processing at SRS

Scenario	*Frequency (per year)*
Explosion: ion exchange column	0.0001
Nuclear criticality [a]	0.0001
Fire	0.000061
Earthquake (design-basis earthquake)	0.00013
Aircraft crash	less than 10^{-7}

[a] Only plutonium criticalities were evaluated. The potential for an americium criticality was considered but dismissed because of the limited americium mass and purity.

- *Explosion*—An explosion in an ion exchange column in the FB-Line was postulated to result from a strong exothermic reaction between nitric acid and the base resin in the cation (or anion) exchange column during plutonium solution exchange. This would result in a thermally induced pressure failure of the ion exchange vessel, and the resulting shrapnel would damage the product run tank and the product hold tank for this ion exchange pair. The explosion would breach the hot cell confinement. The plutonium in nitrite solution in the run and hold tanks would spill onto the cabinet floor and boil due to a subsequent resin fire. Based on the assumptions that the column was at its maximum load before the explosion and that the maximum quantity of liquid at the maximum allowable concentration was present, the estimated release of plutonium through the sand filter and the stack was calculated to be 0.241 grams. No other source term is applicable to the FB-Line accident. Processing in the F-Canyon would remove all other fission products before the plutonium is processed in the FB-Line (DOE 1993b). The frequency of such an event is estimated to be 0.0001 per year.

- *Fire*—In the F-Canyon Safety Analysis Report, a maximum fire was postulated to occur in the plutonium recycle process. The frequency of such a fire was estimated at 0.000061 per year (WSRC 1994). The accident was assumed to burn the contents of the largest tank. The material at risk is 86,700 kilograms (191,000 pounds) of solution. The airborne release fraction and respirable fraction were each estimated to be 0.01 (DOE 1994b). The airborne materials would pass through a sand filter, with a leak path factor of 0.005, before entering the atmosphere. The maximum recycle fire in the F-Canyon would result in the bounding source term (**Table F–14** gives the source terms). Fire in the FB-Line would result in consequences that are several times lower than those from the F-Canyon fire.

- *Nuclear criticality*—A plutonium solution criticality was postulated. The nuclear criticality was assumed to consist of an initial burst of 1×10^{18} fissions in 0.5 seconds, followed at 10-minute intervals for the next 8 hours by bursts of 2×10^{17} fissions, for a total of 1×10^{19} fissions, as specified in the U.S. Nuclear Regulatory Commission's Regulatory Guide 3.35 (NRC 1979) and NUREG-1320 (NRC 1988) and in the DOE-HDBK-3010-YR report (DOE 1994b). The 10^{19} fission yield was based on the assumptions that the solution criticality occurred in a tank with a minimum volume of 3,785 liters (1,000 gallons) and that approximately 100 liters (26 gallons) of this volume evaporated due to heat released during the fission process. Based on the data provided in the DOE Safety Survey Report (DOE 1993c), a 10^{19} criticality event in the FB-Line process would result in the bounding source term (**Table F–15** gives the source terms). The frequency of such an event was estimated to be 0.0001 per year.

Table F–14 Maximum Fire Source Terms

Isotope	*Source Term (curies)*
Sr-90	1.5
Ru-106	12
Ce-144	36
U-234	3.0×10^{-7}
U-235	4.8×10^{-6}
U-236	4.9×10^{-6}
U-238	0.00044
Pu-238	0.19
Pu-239	1.6
Pu-240	0.36
Pu-241	4.2
Pu-242	0.000053
Am-241	0.32

Source: WSRC 1994.

Table F–15 Criticality Source Terms for 10^{19} Fissions in Plutonium Solution

Isotope	*Radioactivity (curies)* [a]			*Airborne Release Fraction* [b]	*Leak Path Factor* [c]	*Source Term (curies)*
	0 to 30 Minutes	*30 Minutes to 8 Hours*	*Total*			
Kr-83m	15	95	110	1	1	110
Kr-85m	9.9	61	70.9	1	1	70.9
Kr-85	0.00012	0.00072	0.00084	1	1	0.00084
Kr-87	60	370	430	1	1	430
Kr-88	32	200	232	1	1	232
Kr-89	1,800	11,000	12,800	1	1	12,800
Xe-131m	0.014	0.086	0.1	1	1	0.1
Xe-133m	0.31	1.9	2.21	1	1	2.21
Xe-133	3.8	23	26.8	1	1	26.8
Xe-135m	460	2,800	3,260	1	1	3,260
Xe-135	57	350	407	1	1	407
Xe-137	6,900	42,000	48,900	1	1	48,900
Xe-138	1,500	9,500	11,000	1	1	11,000
I-131	1.5	9.5	11	0.25	1	2.75
I-132	170	1,000	1,170	0.25	1	293
I-133	22	140	162	0.25	1	40.5
I-134	600	3,700	4,300	0.25	1	1,075
I-135	63	390	453	0.25	1	113
Pu-238 [c,d]			3.6	0.0005	0.005	9.0×10^{-6}
Pu-239 [c,d]			170	0.0005	0.005	0.00043
Pu-240 [c,d]			39	0.0005	0.005	0.0001
Pu-241 [c,d]			2,400	0.0005	0.005	0.006
Pu-242 [c,d]			0.003	0.0005	0.005	7.50×10^{-9}

[a] Regulatory Guide 3.35 (NRC 1979).
[b] Airborne release fractions are equal to 1 for noble gases, 0.25 for iodine, and 0.0005 for plutonium; all particles were assumed to be in the respirable range (i.e., respirable fraction = 1).
[c] Plutonium in 100 liters of solution.
[d] This plutonium was assumed to be released to the atmosphere through a high-efficiency particulate air filter (e.g., SRS's sand filter) with a 0.995 efficiency. The plutonium values are the maximum solution concentration in the FB-Line (DOE 1993b).

- *Earthquake*—Recent analyses of earthquake hazards at F-Canyon indicate that a 0.24 g peak ground acceleration-level earthquake—with a return period of 8,000 years (or a frequency of 0.000125 per year) for the F-Canyon facility—could damage the structure and cause localized interior failures as well as interior and exterior wall cracks (DOE 1996b). Previous analyses of earthquake hazards at F-Canyon estimated the consequences of such an earthquake magnitude with a higher frequency of occurrences—0.0002 per year (DOE 1995b and WSRC 1994). Using the assumptions in the F-Canyon Facility Safety Analysis Report (WSRC 1994), a bounding source term was developed for an earthquake accident (**Table F–16** gives the F-Canyon source terms). Given an earthquake, it was assumed that the plutonium contents in all the processes (F-Canyon and FB-Line) would be spilled on the canyon floor. It was assumed further that the airborne material would enter the environment through the building cracks, which are formed by the loss of sealant between the sections because of differential motion of the section, with a penetration leak path factor of 0.10. For the FB-Line, the material at risk was assumed to be 2,000 grams (4.4 pounds) of plutonium in a molten metal form and 2,000 grams (4.4 pounds) of plutonium in a liquid form. The airborne release fraction multiplied by the respirable fraction is 0.0022 for the molten metal form and 0.000047 for the liquid form, including both the initial and resuspended airborne release fraction multiplied by respirable fraction values. This results in an FB-Line earthquake source term of 0.45 grams of plutonium released to the environment.

Table F–16 Maximum Earthquake Source Terms

Isotope	*Source Term (curies)*	*Isotope*	*Source Term (curies)*
Sr-90	0.086	Pu-239	0.092
Ru-106	70.1	Pu-240	0.021
Ce-144	2.05	Pu-241	0.24
Cs-137	0.0029	Pu-242	3.87×10^{-6}
Eu-154	0.017	Am-241	0.0092
Np-237	2.92×10^{-8}	Am-242m	0.000032
Np-239	0.0058	Am-243	0.0031
U-234	2.06×10^{-7}	Cm-244	0.33
U-235	2.79×10^{-7}	Cm-245	0.000027
U-236	2.81×10^{-7}	Cm-246	0.000042
U-238	0.000025	Cm-247	2.05×10^{-10}
Pu-238	0.015	—	—

Source: WSRC 1994.

- *Aircraft crash*—The F-Canyon facility is located more than 40 kilometers (25 miles) away from any major airport; therefore, no takeoff or landing crash accidents need to be considered. The crashes that could occur in flight need to be considered. According to the DOE Standard on aircraft crash analysis (DOE 1996c), the expected crash frequency for the site is approximately 0.00052 per square kilometer (0.0002 per square mile) per year from general aviation; 1.56×10^{-6} and 5.18×10^{-6} per square kilometer (6×10^{-7} and 2×10^{-6} per square mile) per year from air carrier and air taxies, respectively; and 2.59×10^{-7} and 1.56×10^{-6} per square kilometer (1×10^{-7} and 6×10^{-7} per square mile) per year from large military and small military aircraft, respectively. Using the building dimensions and the data provided in the DOE Standard for aircraft crash analysis, an upper-bound frequency for an aircraft crash into the canyon buildings was estimated to be 4.6×10^{-6} and 1.5×10^{-7} per year for general aviation and commuter (air taxi) aircraft, respectively. These values were calculated without considering any site-specific effects (e.g., the topography and building structures around the facility). Considering the available skid distance of 60 meters (200 feet) that an aircraft could skid before hitting the building, the frequency of an air taxi crashing into the building would be less than 10^{-8} per year. When only crashes that directly hit the structure were considered, general aviation aircraft would have the only estimated crash frequency greater than 10^{-7} per year. The F-Canyon building is a maximum-resistant construction structure designed to withstand a pressure of 47.9 kilopascal (1,000 pounds per square foot). Therefore, crashes of small aircraft (helicopter or a small observation/security aircraft) into

these buildings are not expected to damage the buildings. If a general aviation aircraft were to crash into the buildings, its consequences (both in magnitude and frequency) would be smaller than that hypothesized for a design-basis earthquake.

Accident Scenario Descriptions for the Melt and Dilute Process—The following accidents were considered for the melt and dilute option, when performed at Building 105-L (after receipt of the declad and cleaned spent nuclear fuel at the L-Reactor Disassembly Basin), as proposed in the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000). In this process, the declad and cleaned blanket spent nuclear fuel, along with aluminum metal, would be heated to approximately 1,000 °C (1,830 °F) to form an alloy of 30 percent uranium and 70 percent aluminum, and would be cast as ingots. The heating process would remove some of the radionuclides found in the spent nuclear fuel. The analysis assumed a batch size of 60 kilograms (132 pounds) of heavy metal, which is the batch size limit for this process when performed in Building 105-L. This would lead to three years of operations to melt and dilute the blanket fuel. The radionuclide content of an EBR-II radial blanket spent nuclear fuel batch was used conservatively to represent the radionuclide content of all blanket spent nuclear fuel. The accident scenarios identified in **Table F–17**, and described in the following paragraphs, are applicable to the melt and dilute processing of the blanket spent nuclear fuel in SRS Building 105-L. Accidents associated with the onsite transfer and storage of the declad and cleaned spent nuclear fuel were considered for analysis. As in the accident analysis for the PUREX process, these accidents were not quantified. Accidents associated with the transfer and storage of the spent nuclear fuel and diluted waste forms were assessed as having no additional impacts beyond those analyzed for process-related accidents.

Table F–17 Selected Accident Scenarios for Melt and Dilute Processing at SRS Building 105-L

Scenario	*Frequency (per year)*
Melter eruption/explosion [a]	0.0005
Waste handling spill	0.0064
Loss of electric power	0.006
Fire	0.075
Design-basis earthquake	Not applicable [b]

[a] In the draft EIS, this accident was identified as "loss of cooling water." Consistent with the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000), the accident name was changed.

[b] Building 105-L and the melt and dilute components are expected to remain functioning after a design-basis earthquake. The most significant impact of this event would be a potential loss of offsite power. The consequences of an earthquake up to a design-basis level thereby would be bounded by the loss-of-power event. The loss-of-power event has a higher frequency than the design-basis earthquake and is used in place of the design-basis earthquake.

- *Melter eruption/explosion*—The postulated melter eruption/explosion event could result from a buildup or addition of impurities to the metal melt. Impurities range from water (causing a steam explosion) to chemical contaminants (possible high-temperature exothermic reactions). As a result of the reaction in the metal melt, molten material would be ejected from the melter into the processing structure. Cooling water pipes within the process area could be ruptured as a result of contact with the ejected material. Should this occur, the water released would be converted to steam, resulting in an overpressurization of the enclosure that would be expected to overwhelm the exhaust fans, causing a failure of the exhaust system and an unfiltered release. Although some damage to the exhaust system is expected, there would be insufficient energy in the explosion to damage the facility structure. The melter eruption was assumed to occur with a coincident failure of the high-efficiency particulate air filtration system. The frequency of this event has been estimated to be bound by a value of 0.0005 per year (DOE 2000).

  The material at risk was estimated conservatively to be the full radionuclide content of one melt batch. The metal melt eruption/explosion was assumed to affect all the material in the melter, resulting in a damage ratio of 1 for all material. The airborne release fraction and respirable fraction values were each estimated to be 0.001 for all airborne particulates except cesium, which was estimated to be 0.2 (WSRC 2000, DOE 2000). After such an accident, the particulates would be released in the building and the ventilation fan would draw

the airborne particulates to the building stack. Since the ventilation system was assumed to have failed, the leak path factor was assumed to be 1, allowing all the airborne particulates to enter the environment through the building stack. The material at risk and release fractions are summarized in **Table F–18**.

Table F–18 Melt and Dilute Process Material At Risk and Release Fraction Values for a Melter Eruption/Explosion at Building 105-L

Isotope	*Material at Risk (curies)*	*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Source Term (curies)*
Sr-90	48.4	1.0	0.001	1.0	1.0	0.048
Y-90	48.4	1.0	0.001	1.0	1.0	0.048
Ru-106	8.1	1.0	0.001	1.0	1.0	0.0081
Rh-106	8.1	1.0	0.001	1.0	1.0	0.0081
Cd-113M	0.0427	1.0	0.001	1.0	1.0	0.00043
Sb-125	1.39	1.0	0.001	1.0	1.0	0.0014
Te-125M	0.571	1.0	0.081	1.0	1.0	0.0057
I-129	0.000086	1.0	1	1.0	1.0	0.000086
Cs-134	0.804	1.0	0.2	1.0	1.0	0.1608
Cs-137	104	1.0	0.2	1.0	1.0	20.8
Ba-137M	98.4	1.0	0.001	1.0	1.0	0.0984
Ce-144	3.76	1.0	0.001	1.0	1.0	0.00376
Pr-144	3.76	1.0	0.001	1.0	1.0	0.00376
Pm-147	24.4	1.0	0.001	1.0	1.0	0.0244
Sm-151	6	1.0	0.001	1.0	1.0	0.006
Eu-154	0.44	1.0	0.001	1.0	1.0	0.00044
Eu-155	2.89	1.0	0.001	1.0	1.0	0.00289
Th-228	9.30×10^{-6}	1.0	0.001	1.0	1.0	9.3×10^{-9}
U-234	0.00008	1.0	0.001	1.0	1.0	8×10^{-8}
U-235	0.000226	1.0	0.001	1.0	1.0	2.26×10^{-7}
U-236	0.000254	1.0	0.001	1.0	1.0	2.54×10^{-7}
U-238	0.0196	1.0	0.001	1.0	1.0	1.96×10^{-5}
Np-237	7.80×10^{-6}	1.0	0.001	1.0	1.0	7.80×10^{-9}
Pu-238	0.563	1.0	0.001	1.0	1.0	0.000563
Pu-239	45.2	1.0	0.001	1.0	1.0	0.0452
Pu-240	3.11	1.0	0.001	1.0	1.0	0.00311
Pu-241	12.6	1.0	0.001	1.0	1.0	0.0126
Am-241	0.978	1.0	0.001	1.0	1.0	0.000978
Am-242M	0.0101	1.0	0.001	1.0	1.0	0.000101

- *Waste handling accident*—The filters used in the melt and dilute off-gas exhaust system must be periodically cleaned and the resultant liquid waste disposed of. Decontamination of the filters was assumed to be performed after 10 batches are processed. Therefore, it was assumed that after processing 600 kilograms (1,320 pounds) of heavy metal of blanket spent nuclear fuel, the filters would be decontaminated. It was postulated that a spill would occur during the transfer of the decontaminant liquid from one container to another. The event frequency is estimated at 0.0024 per year (DOE 2000). The material at risk is from the fission products released during the melting process and collected on the filters. This includes fission products with boiling points at or below 1,000 °C (1,830 °F) and some metal oxides that can be expected to form during the heating process (WSRC 1998b). A damage ratio of 0.5 was assumed to account for the spilling of half of the material during the accident. Airborne release fraction and respirable fraction values of 0.0002 and 0.5, respectively, were chosen for the material based on the release of material from aqueous spills (DOE 1994b). The spill was assumed to occur in an area not provided with a filtration system and, therefore, the leak path factor is 1. The material at risk, release fractions, and curies released for this accident for EBR-II blanket spent nuclear fuel are presented in **Table F–19**.

Table F–19 Melt and Dilute Process Material At Risk and Release Fraction Values for a Waste Handling Accident at Building 105-L

Isotope	*Material at Risk*	*Damage Ratio* [a]	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Source Term (curies)*
Te-125M	5.71	0.5	0.0002	0.5	1	0.000286
I-129	0.000864	0.5	0.0002	0.5	1	4.32×10^{-8}
Cs-134	8.04	0.5	0.0002	0.5	1	0.000402
Cs-137	1040	0.5	0.0002	0.5	1	0.052
Pu-238	5.63	0.000015	0.0002	0.5	1	8.45×10^{-9}
Pu-239	452	0.000015	0.0002	0.5	1	6.78×10^{-7}
Pu-240	31.1	0.000015	0.0002	0.5	1	4.67×10^{-8}
Pu-241	126	0.000015	0.0002	0.5	1	1.89×10^{-7}
Am-241	9.78	0.000015	0.0002	0.5	1	1.47×10^{-8}
Am-242M	0.101	0.000015	0.0002	0.5	1	1.52×10^{-10}

[a] Damage ratios for neptunium, plutonium, and americium include an airborne release fraction value of 0.00003 to account for the fraction released from liquid metals and captured on the filters.

- *Loss of offsite power*—The loss of offsite power, with the subsequent failure of the onsite power supply, would result in the failure of the off-gas system, and a potential unfiltered release path to the environment. The probability of this combination of events was conservatively estimated at 0.006 per year (WSRC 1998a). The material at risk was assumed to be the volatile radionuclide inventory of one processing batch of material (approximately 60 kilograms [132 pounds] of heavy metal). Additionally, some amount of radioactive metallic and metallic oxide dusts could be generated and released during a loss-of-power event. The airborne release fraction and respirable fraction values for the gaseous fission products were each assumed to be 1, while the metallic dust release fractions at elevated temperatures are an airborne release fraction of 0.00003 and a respirable fraction of 0.04 (DOE 1994b). A leak path factor of 0.5 has been used for all material to account for possible plate-out during migration of material out of the processing area. The material at risk and release fraction values are summarized in **Table F–20**.

Table F–20 Melt and Dilute Process Material At Risk and Release Fraction Values for a Loss-of-Power Event at Building 105-L

Isotope	*Material at Risk*	*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Source Term (curies)*
Te-125M	0.571	1	1	1	0.5	0.0286
I-129	0.000086	1	1	1	0.5	0.0000432
Cs-134	0.804	1	1	1	0.5	0.0402
Cs-137	104	1	1	1	0.5	52
Pu-238	0.563	1	0.00003	0.04	0.5	3.38×10^{-7}
Pu-239	45.2	1	0.00003	0.04	0.5	0.0000271
Pu-240	3.11	1	0.00003	0.04	0.5	1.87×10^{-6}
Pu-241	12.6	1	0.00003	0.04	0.5	7.56×10^{-6}
Am-241	0.978	1	0.00003	0.04	0.5	5.87×10^{-7}
Am-242M	0.0101	1	0.00003	0.04	0.5	6.06×10^{-9}

- *Area fire*—Fires in Building 105-L have the potential to release material from the chemical decontaminate solution and the off-gas filters and baffles, and have the potential to affect the ventilation and filtration system, resulting in the release modeled for the loss-of-power event. The fire selected for analysis would result in the failure of the waste container and would release some of the decontaminate solution. This fire would have the potential to release more material than a fire that impacts the off-gas filters and baffles. The frequency of a fire in Building 105-L, based on site-wide fire data for SRS, is 0.075 fires per year. This

frequency has been conservatively used as the frequency of a fire that impacts the chemical decontaminate solution. The material at risk would be the same as for the waste handling accident—the volatile gases and metallic and metallic oxide dust that would result from processing 10 batches of material in the melter. All material in the waste container would be at risk and the damage ratio was assumed to be 1. Boiling of a shallow pool of aqueous solution would result in bounding airborne release fraction and respirable fraction values of 0.002 and 1, respectively (DOE 1994b). No credit was taken for any reduction due to the leak path factor (i.e., a leak path factor of 1 was used). **Table F–21** summarizes the material at risk and release fractions for this accident scenario.

Table F–21 Melt and Dilute Process Material At Risk and Release Fraction Values for an Area Fire at Building 105-L

Isotope	*Material at Risk*	*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Source Term (curies)*
Te-125M	5.71	1	0.002	1	1	0.0114
I-129	0.00086	1	0.002	1	1	1.73×10^{-6}
Cs-134	8.04	1	0.002	1	1	0.0161
Cs-137	1040	1	0.002	1	1	2.08
Pu-238	5.63	0.00003	0.002	1	1	3.38×10^{-7}
Pu-239	452	0.00003	0.002	1	1	0.0000271
Pu-240	31.1	0.00003	0.002	1	1	1.87×10^{-6}
Pu-241	126	0.00003	0.002	1	1	7.56×10^{-6}
Am-241	9.78	0.00003	0.002	1	1	5.87×10^{-7}
Am-242M	0.101	0.00003	0.002	1	1	6.06×10^{-9}

F.2.2.2 Consequences and Risk Calculations

Once the source term for each accident scenario is determined, the radiological consequences are calculated. The calculations vary depending on how the release is dispersed, what material is involved, and which receptor is being considered. Risks are calculated based on the accident's frequency and its consequences. The risks also are stated in terms of additional cancer fatalities resulting from a release, using a conversion factor of 0.0005 latent cancer fatalities per person-rem for members of the public and 0.0004 latent cancer fatalities per person-rem for workers.

Radiological consequences to four different receptors were evaluated: a maximally exposed offsite individual (an individual member of the public), the general population, a noninvolved worker, and an involved (facility) worker. The consequences to the facility workers were qualitatively evaluated. For the other receptors, quantitative estimates of consequences were made. Two types of dispersion conditions were considered—95th percentile and 50th 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 that produce higher calculated exposures; it is defined as that condition not exceeded more than 5 percent of the time. Both dispersion conditions were modeled using the GENII program, which determines the desired condition from the site-specific meteorological data in the form of a joint frequency distribution. Joint frequency data are usually produced from at least three consecutive years of site weather data in terms of percentage of time that the wind blows in specific directions (e.g., south, south-southwest, southwest) for the given midpoint (or average) wind speed class and atmospheric stability.

Radiological consequences to a receptor from an accident in the FB-Line were estimated based on a calculated 50-year committed dose factor (dose factor) resulting from releases of 1 gram of plutonium with an isotopic distribution associated with the EBR-II blanket spent nuclear fuel (**Table F–22**). This was done because the FB-Line processes only plutonium already separated in the F-Canyon.

The values given in this table represent the maximum dose to the receptor and were obtained using the GENII program.

Table F–22 Receptors' Dose Factors for Accidental Releases of 1 Gram of Plutonium From an Accident Initiated in the FB-Line

Receptor		*95th Percentile Meteorological Condition*	*50th Percentile Meteorological Condition*
Maximally exposed offsite individual (rem)	Elevated release	0.027	Not applicable
	Ground release	0.13	Not applicable
Population (person-rem)	Elevated release	1500	220
	Ground release	5000	270
Noninvolved worker (rem)	Elevated release	Not applicable	0.080
	Ground release	Not applicable	2

Consequences to involved workers were qualitatively assessed. This approach was used for two reasons: first, no adequate method exists for calculating meaningful consequences at or near the location where the accident could occur. Second, safety assurance for workers is demonstrated by both the workers' training and by the establishment of an Occupational Safety and Health Administration process safety management system (29 CFR 1910.119), the evaluations required by such a system, and the products derived from such evaluations (e.g., procedures, programs, emergency plans).

The consequences to the involved worker, presented in **Tables F-23** and **F–24**, are accident-dependent and site-specific. In facilities where the involved worker activities include remote operations, the consequences of accidents would be lower than in facilities where the workers are near the process. The following paragraphs summarize the various potential consequences to the involved workers from the hypothesized accidents at different sites. Additionally, a limited number of fatalities could occur in an indirect or secondary manner—for example, the involved worker could be killed by an earthquake or explosion.

Table F–23 Involved Worker Consequences From Various Hypothesized Accidents

Accident	*Consequences*
Explosion (ion exchange)	Could potentially result in fatal injuries (nonradiological) to the involved worker (SRS only).
Criticality	Could potentially result in a fatal dose to the involved worker. (Worker location outside cells, e.g., outside the argon cell at ANL-W, provides worker protection.)
Fire	No fatality is expected; some workers could inhale the dispersed radioactive materials before using a respirator and leaving the area.
Earthquake	No fatality is expected.
Spill	Involved workers could inhale the dispersed radioactive materials before using a respirator and leaving the area.

Table F–24 Involved Worker Summary

Accident Description *SRS—PUREX Process*	*Number of Workers at F-Canyon and FB-Line*	*Number of Workers at ANL-W*
Earthquake	47	50
Explosion, ion exchange column	16	Not applicable
Nuclear criticality	16	15
Fire	16	4

- *Explosion*—An explosion could result in serious, even fatal, injuries to involved workers from the accident itself. Some of the involved workers could inhale the dispersed radioactive material before using their respirators and evacuating the area. No fatality is expected from the radiological consequences.

- *Fire*—Involved workers could inhale some radioactive material before evacuating the area. No fatality is expected from the radiological consequences.

- *Spill*—Depending on the location of the spill, nearby workers could inhale the airborne radioactive materials before evacuating the area. Involved workers normally would be wearing respirators when handling the radioactive material containers. No fatality is expected to result from such an accident.

- *Earthquake*—Involved workers could receive lethal injuries from the accident itself. No fatality is expected from radiological consequences.

- *Aircraft Crash*—Consequences similar to those of an earthquake could result from the accident.

- *Criticality*—Involved workers could receive substantial, or potentially fatal, doses from prompt neutrons and gamma rays emitted from the first pulse. After the initial pulse, the workers would evacuate the area immediately on the initiation of the criticality monitoring alarms.

Analysis Conservatism and Uncertainty

To assist in evaluating the impacts of the processing options at SRS and ANL-W on a common basis, a spectrum of generic accidents was postulated for each process location. The accident scenarios were based on similar accidents documented in various site documents. When required, accident assumptions were modified to enable comparison between the sites. In cases where similar accidents were evaluated in site-specific documents, the more conservative analysis assumptions were used for all sites to normalize the results for the purpose of comparison. The following accident analysis parameters have a major impact on accident consequence estimates (i.e., the doses to workers and the public): weather conditions existing at the time of the accident, the material at risk, the isotopic breakdown of the material at risk, and the source term released to the environment.

Weather conditions assumed at the time of the accident have a large impact on dose estimates. Accident impacts to the public (both the maximally exposed offsite individual and the population) presented in this appendix were estimated using both 95th percentile and median 50th percentile weather conditions. The impacts presented in the body of the EIS are based on the 50th percentile weather conditions for the population dose (NRC 1976) and 95th percentile weather conditions (NRC 1983) for the maximally exposed offsite individual dose (which provides conservative maximally exposed offsite individual dose estimates). The GENII computer code was used to calculate doses to the public within 80 kilometers (50 miles) of the accident release point. The code calculates the public dose in each of 16 sectors centered at the accident release point. The GENII computer code also assumes that the total source term is released into each sector and that there is no change in the weather (i.e., wind direction, wind speed, and stability class) while the accident plume is traversing the 80-kilometer sector. The use of the 95th percentile weather data rather than the expected or median 50th percentile weather data was considered to be unrealistic for estimating the population dose. Meteorological conditions used in the analysis are based on measured weather data at the site. The 95th percentile represents a very stable site meteorological condition, which cannot be expected to be applicable for a wide area up to 80 kilometers from the site. Therefore, the 50th percentile, which represents a more neutral weather condition, is more representative of expected weather conditions over a wide area.

Uncertainties in accident frequencies do not impact the accident consequences, but do impact accident risk. The site/facility-specific accident frequencies (i.e., earthquake-induced building damage and aircraft crash) were based on data provided by the sites. Process-specific accident frequencies were estimated based on analyses provided in site-specific documentation. In cases where similar accidents were evaluated in site-specific

documents, the more conservative accident frequency was used for all sites to normalize the results for the purpose of comparison.

Due to the layers of conservatism built into the accident analysis for the spectrum of postulated accidents, the estimated consequences and risks to the public represent the upper limit for the individual classes of accidents. The uncertainties associated with the accident frequency estimates are enveloped by the analysis conservatism.

F.2.3 Accident Analyses Consequences and Risk Results

F.2.3.1 No Action Alternative

Under the No Action Alternative, the sodium-bonded spent nuclear fuel would not be treated (no sodium would be removed from the interior of the fuel elements) except for stabilization activities that may be necessary for continued safe and secure storage until 2035 or until a new treatment technology is developed. Under the Electrometallurgical Treatment Research and Demonstration Project, approximately 0.4 metric tons of heavy metal of EBR-II driver spent nuclear fuel and 1.2 metric tons of heavy metal of blanket spent nuclear fuel were processed. This EIS evaluates the impacts associated with activities required to clean up and stabilize any residual waste materials generated during the demonstration project at ANL-W. In addition, at the completion of the project, any remaining sodium-bonded spent nuclear fuel in the process facilities would be packaged and transferred to dry storage in the Radioactive Scrap and Waste Facility. Spent nuclear fuel transfer activities and waste processing activities would be completed in about two years after equipment installation. Some of the spent nuclear fuel handling and processing accidents identified under Alternative 1 are applicable to the No Action Alternative. **Tables F-25** and **F-26** provide the dose calculation results for the design-basis and beyond-design-basis earthquakes for stabilizing the residual waste. The results for the remaining accidents considered for the No Action Alternative (the salt powder spill in the Hot Fuel Examination Facility, the cask drop, and the transuranic waste fire) are provided in the discussion of "Alternative 1: Electrometallurgically Treat Blanket and Driver Fuel at ANL-W." Consequence and risk results are provided for the maximally exposed offsite individual, a noninvolved worker, and the general population. The accident assumptions and parameters used in developing these results are provided in Section F.2.2 of this appendix. EBR-II driver spent nuclear fuel characteristics (radionuclide compositions), which bound the consequences, were used to represent the consequences and risks during stabilization of waste for the demonstration project for the No Action Alternative. The transuranic waste fire accident was analyzed using a generic transuranic waste package composition.

Table F–25 Summary of Dose Calculation Results for the Design-Basis Earthquake (Driver)

			95th Percentile Meteorology			*50th Percentile Meteorology*			
Accident	*Frequency (event per year)*	*Risk*	*MEI (millirem)*	*Population (person-rem)*	*Average Individual (millirem)*	*MEI (millirem)*	*Noninvolved Worker (millirem)*	*Population (person-rem)*	*Average Individual (millirem)*
Design-basis earthquake	0.008	Dose per event	12	52	0.63	0.64	4.7	1.4	0.017
		Dose per year	0.095	0.42	0.005	0.0051	0.038	0.011	0.00014
		LCF per year	4.8×10^{-8}	0.00021	2.5×10^{-9}	2.6×10^{-9}	1.5×10^{-8}	5.6×10^{-6}	6.8×10^{-8}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality

Table F–26 Summary of Dose Calculation Results for the Beyond-Design-Basis Earthquake (Driver)

			95th Percentile Meteorology			50th Percentile Meteorology			
Accident	*Frequency (event per year)*	*Risk*	*MEI (millirem)*	*Population (person-rem)*	*Average Individual (millirem)*	*MEI (millirem)*	*Noninvolved Worker (millirem)*	*Population (person-rem)*	*Average Individual (millirem)*
Beyond-design- basis earthquake [a]	0.00001	Dose per event	96	42	5.1	5.1	37	11	0.13
		Dose per year	0.00096	0.00042	0.000051	0.000051	0.00037	0.00011	1.3×10^{-6}
		LCF per year	4.8×10^{-10}	2.1×10^{-7}	2.6×10^{-11}	2.6×10^{-11}	1.5×10^{-10}	5.5×10^{-8}	6.5×10^{-13}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.
[a] During stabilization of the demonstration project waste, only the Hot Fuel Examination Facility salt powder spill would be applicable.

F.2.3.2 Alternative 1: Electrometallurgically Treat Blanket and Driver Fuel at ANL-W

The processing technology considered for this alternative consists solely of the electrometallurgical treatment processing of the sodium-bonded spent nuclear fuel at ANL-W, using the Fuel Conditioning Facility and the Hot Fuel Examination Facility. **Tables F–27** through **F–37** provide the dose calculation results for the electrometallurgical treatment-related accidents at ANL-W. Consequence and risk results are provided for the maximally exposed offsite individual, a noninvolved worker, and the general population. The accident assumptions and parameters used in developing these results are provided in Section F.2.2 of this appendix. EBR-II driver and blanket spent nuclear fuel characteristics (radionuclide compositions) were used to develop the consequence and risk factors for all driver and blanket spent nuclear assembly fuel. The transuranic waste fire accident was analyzed using a generic transuranic waste package composition, rather than either a blanket or driver spent nuclear fuel-specific composition.

Table F–27 Summary of Dose Calculation Results for a Salt Powder Spill (Driver)

			95th Percentile Meteorology			50th Percentile Meteorology			
Accident	*Frequency (event per year)*	*Risk*	*MEI (millirem)*	*Population (person-rem)*	*Average Individual (millirem)*	*MEI (millirem)*	*Noninvolved Worker (millirem)*	*Population (person-rem)*	*Average Individual (millirem)*
Hot Fuel Examination Facility salt powder spill	0.01	Dose per event	0.00046	0.0026	0.000031	0.000046	4.7×10^{-7}	0.000098	1.2×10^{-6}
		Dose per year	4.6×10^{-6}	0.000026	3.1×10^{-7}	4.6×10^{-7}	4.7×10^{-9}	9.8×10^{-7}	1.2×10^{-8}
		LCF per year	2.3×10^{-12}	1.3×10^{-8}	1.6×10^{-13}	2.3×10^{-13}	1.9×10^{-15}	4.9×10^{-10}	5.9×10^{-15}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–28 Summary of Dose Calculation Results for a Salt Powder Spill (Blanket)

			95th Percentile Meteorology			50th Percentile Meteorology			
Accident	Frequency (event per year)	Risk	MEI (millirem)	Population (person-rem)	Average Individual (millirem)	MEI (millirem)	Noninvolved Worker (millirem)	Population (person-rem)	Average Individual (millirem)
Hot Fuel Examination Facility salt powder spill	0.01	Dose per event	0.00015	0.00088	0.000011	0.000015	1.3×10^{-6}	0.000033	4.0×10^{-7}
		Dose per year	1.5×10^{-6}	8.8×10^{-6}	1.1×10^{-7}	1.5×10^{-7}	1.3×10^{-8}	3.3×10^{-7}	4.0×10^{-9}
		LCF per year	7.5×10^{-13}	4.4×10^{-9}	5.5×10^{-14}	7.5×10^{-14}	5.3×10^{-15}	1.7×10^{-10}	2.0×10^{-15}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–29 Summary of Dose Calculation Results for a Cask Drop (Driver)

			95th Percentile Meteorology			50th Percentile Meteorology			
Accident	Frequency (event per year)	Risk	MEI (millirem)	Population (person-rem)	Average Individual (millirem)	MEI (millirem)	Noninvolved Worker (millirem)	Population (person-rem)	Average Individual (millirem)
Cask drop	0.01	Dose per event	0.03	0.14	0.0017	0.0016	0.00084	0.0035	0.000042
		Dose per year	0.0003	0.0014	0.000017	0.000016	8.4×10^{-6}	0.000035	4.2×10^{-7}
		LCF per year	1.5×10^{-10}	7.0×10^{-7}	8.5×10^{-12}	8.0×10^{-12}	3.4×10^{-12}	1.7×10^{-8}	2.1×10^{-13}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–30 Summary of Dose Calculation Results for a Cask Drop (Blanket)

			95th Percentile Meteorology			50th Percentile Meteorology			
Accident	Frequency (event per year)	Risk	MEI (millirem)	Population (person-rem)	Average Individual (millirem)	MEI (millirem)	Noninvolved Worker (millirem)	Population (person-rem)	Average Individual (millirem)
Cask drop	0.01	Dose per event	0.0024	0.011	0.00013	0.00013	0.000049	0.00028	3.4×10^{-6}
		Dose per year	0.000024	0.00011	1.3×10^{-6}	1.3×10^{-6}	4.9×10^{-7}	2.8×10^{-6}	3.4×10^{-8}
		LCF per year	1.2×10^{-11}	5.5×10^{-8}	6.5×10^{-13}	6.5×10^{-13}	2.0×10^{-13}	1.4×10^{-9}	1.7×10^{-14}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–31 Summary of Dose Calculation Results for a Single-Container Transuranic Waste Fire

			95th Percentile Meteorology			50th Percentile Meteorology			
Accident	Frequency (event per year)	Risk	MEI (millirem)	Population (person-rem)	Average Individual (millirem)	MEI (millirem)	Noninvolved Worker (millirem)	Population (person-rem)	Average Individual (millirem)
Transuranic waste fire	0.001	Dose per event	0.059	0.27	0.0033	0.0032	0.22	0.0071	0.000085
		Dose per year	0.000059	0.00027	3.3×10^{-6}	3.2×10^{-6}	0.00022	7.1×10^{-6}	8.5×10^{-8}
		LCF per year	3.0×10^{-11}	1.4×10^{-7}	1.6×10^{-12}	1.6×10^{-12}	8.8×10^{-11}	3.6×10^{-9}	4.3×10^{-14}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–32 Summary of Dose Calculation Results for a Design-Basis Earthquake (Driver)

			95th Percentile Meteorology			50th Percentile Meteorology			
Accident	Frequency (event per year)	Risk	MEI (millirem)	Population (person-rem)	Average Individual (millirem)	MEI (millirem)	Noninvolved Worker (millirem)	Population (person-rem)	Average Individual (millirem)
Design-basis earthquake	0.0002 (Multi-facility event)	Dose per event	13	70	0.84	0.95	4.7	2.8	0.034
		Dose per year	0.0026	0.014	0.00017	0.00019	0.00084	0.00056	6.8×10^{-6}
		LCF per year	1.3×10^{-9}	7.0×10^{-6}	8.4×10^{-11}	9.5×10^{-11}	3.8×10^{-10}	2.8×10^{-7}	3.4×10^{-12}
	0.008 (HFEF)	Dose per event	12	52	0.63	0.64	4.7	1.4	0.017
		Dose per year	0.095	0.42	0.0050	0.0051	0.037	0.011	0.00013
		LCF per year	4.8×10^{-8}	0.00021	2.5×10^{-9}	2.6×10^{-9}	1.5×10^{-8}	5.6×10^{-6}	6.6×10^{-11}

MEI = Maximally Exposed Offsite Individual, HFEF = Hot Fuel Examination Facility, LCF = Latent Cancer Fatality.

Table F–33 Summary of Dose Calculation Results for a Design-Basis Earthquake (Blanket)

			95th Percentile Meteorology			50th Percentile Meteorology			
Accident	Frequency (event per year)	Risk	MEI (millirem)	Population (person-rem)	Average Individual (millirem)	MEI (millirem)	Noninvolved Worker (millirem)	Population (person-rem)	Average Individual (millirem)
Design-basis earthquake	0.0002 (Multi-facility event)	Dose per event	4.1	18	0.22	0.23	14	0.49	0.006
		Dose per year	0.00081	0.0036	0.000044	0.000046	0.0027	0.00010	1.2×10^{-6}
		LCF per year	4.1×10^{-10}	1.8×10^{-6}	2.2×10^{-11}	2.3×10^{-11}	1.1×10^{-9}	4.9×10^{-8}	6.0×10^{-13}
	0.008 (HFEF)	Dose per event	4.0	18	0.21	0.22	14	0.47	0.0057
		Dose per year	0.032	0.14	0.0017	0.0018	0.11	0.0038	0.000045
		LCF per year	1.6×10^{-8}	0.000072	8.6×10^{-10}	8.8×10^{-10}	4.5×10^{-8}	1.9×10^{-6}	2.3×10^{-11}

MEI = Maximally Exposed Offsite Individual, HFEF = Hot Fuel Examination Facility, LCF = Latent Cancer Fatality.

Table F–34 Summary of Dose Calculation Results for a Salt Transfer Drop (Driver)

			95th Percentile Meteorology			50th Percentile Meteorology			
Accident	Frequency (event per year)	Risk	MEI (millirem)	Population (person-rem)	Average Individual (millirem)	MEI (millirem)	Noninvolved Worker (millirem)	Population (person-rem)	Average Individual (millirem)
Salt transfer drop	1.0×10^{-7}	Dose per event	0.19	0.84	0.01	0.01	0.073	0.022	0.00026
		Dose per year	1.9×10^{-8}	8.4×10^{-8}	1.0×10^{-9}	1.0×10^{-9}	7.3×10^{-9}	2.2×10^{-9}	2.6×10^{-11}
		LCF per year	9.5×10^{-15}	4.2×10^{-11}	5.0×10^{-16}	5.0×10^{-16}	2.9×10^{-15}	1.1×10^{-12}	1.3×10^{-17}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–35 Summary of Dose Calculation Results for a Salt Transfer Drop (Blanket)

			95th Percentile Meteorology			50th Percentile Meteorology			
Accident	Frequency (event per year)	Risk	MEI (millirem)	Population (person-rem)	Average Individual (millirem)	MEI (millirem)	Noninvolved Worker (millirem)	Population (person-rem)	Average Individual (millirem)
Salt transfer drop	1.0×10^{-7}	Dose per event	0.065	0.29	0.0035	0.0036	0.22	0.0077	0.000092
		Dose per year	6.5×10^{-9}	2.9×10^{-8}	3.5×10^{-10}	3.6×10^{-10}	2.2×10^{-8}	7.7×10^{-10}	9.2×10^{-12}
		LCF per year	3.3×10^{-15}	1.5×10^{-11}	1.8×10^{-16}	1.8×10^{-16}	8.8×10^{-15}	3.9×10^{-13}	4.6×10^{-18}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–36 Summary of Dose Calculation Results for a Beyond-Design-Basis Earthquake (Driver)

			95th Percentile Meteorology			50th Percentile Meteorology			
Accident	Frequency (event per year)	Risk	MEI (millirem)	Population (person-rem)	Average Individual (millirem)	MEI (millirem)	Noninvolved Worker (millirem)	Population (person-rem)	Average Individual (millirem)
Beyond-design-basis earthquake	0.00001	Dose per event	22,000	97,000	1,200	1,200	370	2,500	31
		Dose per year	0.22	0.97	0.012	0.012	0.0037	0.025	0.00031
		LCF per year	2.2×10^{-7}	0.00049	5.9×10^{-9}	6.0×10^{-9}	1.5×10^{-9}	0.000013	1.5×10^{-10}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–37 Summary of Dose Calculation Results for a Beyond-Design-Basis Earthquake (Blanket)

			95th Percentile Meteorology			50th Percentile Meteorology			
Accident	Frequency (event per year)	Risk	MEI (millirem)	Population (person-rem)	Average Individual (millirem)	MEI (millirem)	Noninvolved Worker (millirem)	Population (person-rem)	Average Individual (millirem)
Beyond-design-basis earthquake	0.00001	Dose per event	930	4,200	51	50	560	110	1.3
		Dose per year	0.0093	0.042	0.00051	0.00050	0.0056	0.0011	0.000013
		LCF per year	4.7×10^{-9}	0.000021	2.5×10^{-10}	2.5×10^{-10}	2.3×10^{-9}	5.5×10^{-7}	6.5×10^{-12}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

F.2.3.3 Alternative 2: Clean and Package Blanket Fuel in High-Integrity Cans and Electrometallurgically Treat Driver Fuel at ANL-W

The processing technology considered for this alternative consists of cleaning the sodium from blanket spent nuclear fuel and packaging the cleaned blanket spent nuclear fuel in high-integrity cans. The sodium-bonded driver spent nuclear fuel would be processed using the electrometallurgical treatment process. The dose calculation results for this combination of processes at ANL-W are found in Section F.2.3.2 for driver spent nuclear fuel and in F.2.3.4 for blanket spent nuclear fuel. All of the electrometallurgical treatment accidents for the driver spent nuclear fuel are applicable to this process. For the blanket spent nuclear fuel, the sodium fire and the cask handling accident are applicable. The accident assumptions and parameters used in developing these results are provided in Section F.2.2 of this appendix. EBR-II driver spent nuclear fuel and blanket spent nuclear fuel characteristics (radionuclide compositions) were used to develop the consequence and risk factors for all driver and blanket spent nuclear assembly fuel.

F.2.3.4 Alternative 3: Declad and Clean Blanket Fuel and Electrometallurgically Treat Driver Fuel at ANL-W; PUREX Process Blanket Fuel at SRS

The processing technology considered for this alternative consists of decladding and cleaning the sodium-bonded blanket spent nuclear fuel at the Hot Fuel Examination Facility at ANL-W and shipment of this material to SRS for PUREX processing. In this alternative, the sodium-bonded driver spent nuclear fuel would be processed using the electrometallurgical treatment process at ANL-W. **Tables F–38** through **F–44** provide the dose calculation results for accidents during PUREX processing at SRS and for cask drop and sodium fire accidents at ANL-W. The accident assumptions and parameters used in developing these results are provided in Section F.2.2 of this appendix. EBR-II driver and blanket spent nuclear fuel characteristics (radionuclide compositions) were used to develop the consequence and risk factors for all driver and blanket spent nuclear assembly fuel.

Consequence and risk estimates are provided for both processing the blanket spent nuclear fuel material at ANL-W prior to its shipment to SRS and for processing the material at SRS. Analysis results for processing the driver spent nuclear fuel can be found in the discussion for Alternative 1 in Section F.2.3.2.

Table F–38 Summary of Dose Calculation Results for an F-Canyon Fire

			95th Percentile Meteorology		*50th Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Noninvolved Worker (millirem)***	***Population (person-rem)***
F-Canyon fire	0.000061	Dose per event	610	36,000	2,300	5,500
		Dose per year	0.037	2.2	0.14	0.34
		LCF per year	1.9×10^{-8}	0.0011	5.6×10^{-8}	0.00017

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–39 Summary of Dose Calculation Results for an FB-Line Explosion

			95th Percentile Meteorology		*50th Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Noninvolved Worker (millirem)***	***Population (person-rem)***
FB-Line explosion	0.00010	Dose per event	6.5	360	19	53
		Dose per year	0.00065	0.036	0.0019	0.0053
		LCF per year	3.3×10^{-10}	0.000018	7.6×10^{-10}	2.7×10^{-6}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–40 Summary of Dose Calculation Results for an F-Canyon Earthquake

			95th Percentile Meteorology		*50th Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Noninvolved Worker (millirem)***	***Population (person-rem)***
F-Canyon earthquake	0.00013	Dose per event	1,100	38,000	12,000	2,100
		Dose per year	0.14	4.9	1.56	0.27
		LCF per year	7.2×10^{-8}	0.0025	6.2×10^{-7}	0.00014

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–41 Summary of Dose Calculation Results for an FB-Line Earthquake

			95th Percentile Meteorology		50th Percentile Meteorology	
Accident	Frequency (event per year)	Risk	MEI (millirem)	Population (person-rem)	Noninvolved Worker (millirem)	Population (person-rem)
FB-Line earthquake	0.00013	Dose per event	58	2.250	900	120
		Dose per year	0.0075	0.29	0.12	0.016
		LCF per year	3.8×10^{-9}	0.00015	4.7×10^{-8}	7.8×10^{-6}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–42 Summary of Dose Calculation Results for an F-Canyon Criticality Accident

			95th Percentile Meteorology		50th Percentile Meteorology	
Accident	Frequency (event per year)	Risk	MEI (millirem)	Population (person-rem)	Noninvolved Worker (millirem)	Population (person-rem)
F-Canyon criticality	0.00010	Dose per event	11	380	37	59
		Dose per year	0.0011	0.038	0.0037	0.0059
		LCF per year	5.5×10^{-10}	0.000019	1.5×10^{-9}	3.0×10^{-6}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–43 Summary of Dose Calculation Results for an ANL-W Cask Drop Accident

			95th Percentile Meteorology		50th Percentile Meteorology	
Accident	Frequency (event per year)	Risk	MEI (millirem)	Population (person-rem)	Noninvolved Worker (millirem)	Population (person-rem)
Cask drop	0.01	Dose per event	0.0024	0.011	0.000049	0.00028
		Dose per year	0.000024	0.00011	4.9×10^{-7}	2.8×10^{-6}
		LCF per year	1.2×10^{-11}	5.5×10^{-8}	2.0×10^{-13}	1.4×10^{-9}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–44 Summary of Dose Calculation Results for an ANL-W Sodium Fire

			95th Percentile Meteorology		50th Percentile Meteorology	
Accident	Frequency (event per year)	Risk	MEI (millirem)	Population (person-rem)	Noninvolved Worker (millirem)	Population (person-rem)
Sodium fire during decladding and cleaning	0.008	Dose per event	5.9	26.3	0.054	0.69
		Dose per year	0.047	0.21	0.00043	0.0055
		LCF per year	2.4×10^{-8}	0.00011	1.7×10^{-10}	2.8×10^{-6}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

F.2.3.5 Alternative 4: Melt and Dilute Blanket Fuel and Electrometallurgically Treat Driver Fuel at ANL-W

The processing technology considered for this alternative consists of melting and diluting the cleaned blanket spent nuclear fuel at the Hot Fuel Examination Facility at ANL-W. In this alternative, the sodium-bonded driver spent nuclear fuel would be processed using the electrometallurgical treatment process at ANL-W. The dose calculation results for this alternative are provided in this section. The results for the driver spent nuclear fuel are presented as part of the results for Alternative 1 (Section F.2.3.2) and the results for the blanket spent

nuclear fuel are presented as part of the results for Alternative 6 (Section F.2.3.7), where the results for melt and dilute processing of both driver and blanket spent nuclear fuel are presented. The accident assumptions and parameters used in developing these results are provided in Section F.2.2 of this appendix. EBR-II driver and blanket spent nuclear fuel characteristics (radionuclide compositions) were used to develop the consequence and risk factors for all driver and blanket spent nuclear assembly fuel.

F.2.3.6 Alternative 5: Declad and Clean Blanket Fuel and Electrometallurgically Treat Driver Fuel at ANL-W; Melt and Dilute Blanket Fuel at SRS

The processing technology considered for this alternative consists of decladding, cleaning, and packaging the blanket spent nuclear fuel at the Hot Fuel Examination Facility at ANL-W and shipping the packaged blanket spent nuclear fuel to SRS for melt and dilute processing in Building 105-L. In this alternative, the sodium-bonded driver spent nuclear fuel would be processed using the electrometallurgical treatment process at ANL-W. **Tables F–45** through **F–50** provide the dose calculation results for the melt and dilute process at SRS. The accident assumptions and parameters used in developing these results are provided in Section F.2.2 of this appendix. EBR-II driver and blanket spent nuclear fuel characteristics (radionuclide compositions) were used to develop the consequence and risk factors for all driver and blanket spent nuclear assembly fuel.

Consequence and risk estimates are provided for both processing the blanket spent nuclear material at ANL-W prior to its shipment to SRS, and for processing the material at SRS. Analysis results for processing driver spent nuclear fuel can be found in the discussion for Alternative 1 in Section F.2.3.2.

Table F–45 Summary of Dose Calculation Results for an L-Area Waste Handling Accident

			95th Percentile Meteorology		*50th Percentile Meteorology*	
Accident	*Frequency (event per year)*	*Risk*	*MEI (millirem)*	*Population (person-rem)*	*Noninvolved Worker (millirem)*	*Population (person-rem)*
Waste handling spill	0.0064	Dose per event	2.1	42	0.17	3.6
		Dose per year	0.013	0.27	0.0011	0.023
		LCF per year	6.7×10^{-9}	0.000014	5.5×10^{-10}	0.000012

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–46 Summary of Dose Calculation Results for a Building 105-L Loss-of-Power Event

			95th Percentile Meteorology		*50th Percentile Meteorology*	
Accident	*Frequency (event per year)*	*Risk*	*MEI (millirem)*	*Population (person-rem)*	*Noninvolved Worker (millirem)*	*Population (person-rem)*
Loss-of-power event	0.006	Dose per event	2,100	42,000	140	3,500
		Dose per year	12.6	250	0.84	21
		LCF per year	6.3×10^{-6}	0.13	3.4×10^{-7}	0.011

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–47 Summary of Dose Calculation Results for a Building 105-L Melter Eruption/Explosion

			95th Percentile Meteorology		*50th Percentile Meteorology*	
Accident	*Frequency (event per year)*	*Risk*	*MEI (millirem)*	*Population (person-rem)*	*Noninvolved Worker (millirem)*	*Population (person-rem)*
Melter eruption/ explosion	0.0005	Dose per event	269	6,390	72.9	1,160
		Dose per year	0.14	3.2	0.037	0.58
		LCF per year	7.0×10^{-8}	0.0016	1.5×10^{-8}	0.00029

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–48 Summary of Dose Calculation Results for a Building 105-L Fire

			95th Percentile Meteorology		50th Percentile Meteorology	
Accident	*Frequency (event per year)*	*Risk*	*MEI (millirem)*	*Population (person-rem)*	*Noninvolved Worker (millirem)*	*Population (person-rem)*
Fire	0.075	Dose per event	86	1.700	6.3	140
		Dose per year	6.5	130	0.47	11
		LCF per year	3.2×10^{-6}	0.064	1.9×10^{-7}	0.0053

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–49 Summary of Dose Calculation Results for an ANL-W Cask Drop Accident

			95th Percentile Meteorology		50th Percentile Meteorology	
Accident	*Frequency (event per year)*	*Risk*	*MEI (millirem)*	*Population (person-rem)*	*Noninvolved Worker (millirem)*	*Population (person-rem)*
Cask drop	0.01	Dose per event	0.0024	0.011	0.000049	0.00028
		Dose per year	0.000024	0.00011	4.9×10^{-7}	2.8×10^{-6}
		LCF per year	1.2×10^{-11}	5.5×10^{-8}	2.0×10^{-13}	1.4×10^{-9}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–50 Summary of Dose Calculation Results for an ANL-W Sodium Fire

			95th Percentile Meteorology		50th Percentile Meteorology	
Accident	*Frequency (event per year)*	*Risk*	*MEI (millirem)*	*Population (person-rem)*	*Noninvolved Worker (millirem)*	*Population (person-rem)*
Sodium fire during decladding and cleaning	0.008	Dose per event	5.9	26.3	0.054	0.69
		Dose per year	0.047	0.21	0.00043	0.0055
		LCF per year	2.4×10^{-8}	0.00011	1.7×10^{-10}	2.8×10^{-6}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

F.2.3.7 Alternative 6: Melt and Dilute Blanket and Driver Fuel at ANL-W

The processing technology considered for this alternative consists of cleaning both blanket and driver spent nuclear fuel and melting and diluting the spent nuclear fuel at the Hot Fuel Examination Facility at ANL-W. **Tables F–51** through **F–57** provide the dose calculation results for the melt and dilute process at ANL-W. The accident assumptions and parameters used in developing these results are provided in Section F.2.2 of this appendix. EBR-II driver and blanket spent nuclear fuel characteristics (radionuclide compositions) were used to develop the consequence and risk factors for all driver and blanket spent nuclear assembly fuel.

Consequence and risk estimates are provided for both the declad and clean processing and the melt and dilute processing of the sodium-bonded spent nuclear fuel.

Table F–51 Summary of Dose Calculation Results for a Melt and Dilute Design-Basis Event (Driver)

			95th Percentile Meteorology			50th Percentile Meteorology			
Accident	Frequency (event per year)	Risk	MEI (millirem)	Population (person-rem)	Average Individual (millirem)	MEI (millirem)	Noninvolved Worker (millirem)	Population (person-rem)	Average Individual (millirem)
Design-basis earthquake (includes sodium fire)	0.008	Dose per event	19,000	89,400	1,080	1,080	838	2,250	27
		Dose per year	152	715.2	8.64	8.64	6.7	18	0.22
		LCF per year	0.000076	0.36	4.3×10^{-6}	4.3×10^{-6}	2.7×10^{-6}	0.0090	1.1×10^{-7}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–52 Summary of Dose Calculation Results for a Melt and Dilute Design-Basis Event (Blanket)

			95th Percentile Meteorology			50th Percentile Meteorology			
Accident	Frequency (event per year)	Risk	MEI (millirem)	Population (person-rem)	Average Individual (millirem)	MEI (millirem)	Noninvolved Worker (millirem)	Population (person-rem)	Average Individual (millirem)
Design-basis earthquake (includes sodium fire)	0.008	Dose per event	471	2240	26.9	27	15.2	56.1	0.68
		Dose per year	3.8	17.92	0.22	0.22	0.12	0.45	0.0054
		LCF per year	1.9×10^{-6}	0.0090	1.1×10^{-7}	1.1×10^{-7}	4.8×10^{-8}	0.00022	2.7×10^{-9}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–53 Summary of Dose Calculation Results for a Melt and Dilute Waste Handling Accident (Driver)

			95th Percentile Meteorology			50th Percentile Meteorology			
Accident	Frequency (event per year)	Risk	MEI (millirem)	Population (person-rem)	Average Individual (millirem)	MEI (millirem)	Noninvolved Worker (millirem)	Population (person-rem)	Average Individual (millirem)
Waste handling accident (liquid spill)	0.0024	Dose per event	597	2820	34	33.9	26.7	70.8	.85
		Dose per year	1.43	6.77	0.082	0.081	0.064	0.17	0.0020
		LCF per year	7.2×10^{-7}	0.0034	4.1×10^{-8}	4.1×10^{-8}	2.6×10^{-8}	0.000085	1.0×10^{-9}

MEI - Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–54 Summary of Dose Calculation Results for a Melt and Dilute Waste Handling Accident (Blanket)

			95th Percentile Meteorology			50th Percentile Meteorology			
Accident	Frequency (event per year)	Risk	MEI (millirem)	Population (person-rem)	Average Individual (millirem)	MEI (millirem)	Noninvolved Worker (millirem)	Population (person-rem)	Average Individual (millirem)
Waste handling accident (liquid spill)	0.0024	Dose per event	14.9	70.8	0.85	0.85	0.49	1.8	0.022
		Dose per year	0.036	0.17	0.0020	0.0020	0.0012	0.0043	0.000053
		LCF per year	1.8×10^{-8}	0.000085	1.0×10^{-9}	1.0×10^{-9}	4.8×10^{-10}	2.2×10^{-6}	2.7×10^{-11}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–55 Summary of Dose Calculation Results for a Melt and Dilute Criticality Accident (Driver)

			95th Percentile Meteorology			50th Percentile Meteorology			
Accident	Frequency (event per year)	Risk	MEI (millirem)	Population (person-rem)	Average Individual (millirem)	MEI (millirem)	Noninvolved Worker (millirem)	Population (person-rem)	Average Individual (millirem)
Criticality	0.0003	Dose per event	0.52	1.6	0.019	0.083	0.47	0.085	1.0×10^{-6}
		Dose per year	0.00016	0.00048	0.0000057	0.000025	0.00014	0.000026	3.0×10^{-10}
		LCF per year	8.0×10^{-11}	2.4×10^{-7}	2.9×10^{-12}	1.3×10^{-11}	5.6×10^{-11}	1.3×10^{-8}	1.5×10^{-16}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–56 Summary of Dose Calculation Results for a Melt and Dilute Sodium Fire (Driver)

			95th Percentile Meteorology			50th Percentile Meteorology			
Accident	Frequency (event per year)	Risk	MEI (millirem)	Population (person-rem)	Average Individual (millirem)	MEI (millirem)	Noninvolved Worker (millirem)	Population (person-rem)	Average Individual (millirem)
Sodium fire	0.008	Dose per event	282	1,260	15.2	15.6	2.59	33	0.4
		Dose per year	0.23	10.08	0.12	0.12	0.021	0.26	0.0032
		LCF per year	1.13×10^{-6}	0.0050	6.0×10^{-8}	6.0×10^{-8}	8.3×10^{-9}	0.00013	1.6×10^{-9}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

Table F–57 Summary of Dose Calculation Results for a Melt and Dilute Sodium Fire (Blanket)

			95th Percentile Meteorology			50th Percentile Meteorology			
Accident	Frequency (event per year)	Risk	MEI (millirem)	Population (person-rem)	Average Individual (millirem)	MEI (millirem)	Noninvolved Worker (millirem)	Population (person-rem)	Average Individual (millirem)
Sodium fire	0.008	Dose per event	5.9	26.3	0.32	0.33	0.054	0.69	0.0083
		Dose per year	0.047	0.21	0.0026	0.0026	0.00043	0.0055	0.000066
		LCF per year	2.4×10^{-8}	0.00011	1.3×10^{-9}	1.3×10^{-9}	1.7×10^{-10}	2.8×10^{-6}	3.3×10^{-11}

MEI = Maximally Exposed Offsite Individual, LCF = Latent Cancer Fatality.

F.3 Impacts of Hazardous Chemical Accidents on Human Health

F.3.1 Chemical Accident Analysis Methodology

Factors such as receptor location, terrain, meteorological conditions, release conditions, and characteristics of the chemical inventory are required as input parameters for hand calculations or computer codes to determine human exposure from airborne releases of toxic chemicals. This section gives a general narrative about these input parameters with degrees of conservatism noted, and describes the computer models used to perform exposure estimates. EPIcode™ is the computer code chosen for estimating airborne concentrations resulting from most releases of toxic chemicals (Homann 1988).

F.3.1.1 *EPIcode™*

EPIcode™ uses the well-established Gaussian Plume Model to calculate the airborne toxic chemical concentrations at the receptor locations. The EPIcode™ library contains information on over 600 toxic substances listed in the *Threshold Limit Values for Chemical Substances and Physical Agents and Biomedical Exposure Indices* (ACGIH 1994). The types of releases that can be modeled, and associated input parameters, are discussed below.

Continuous release models require specifying the source term as an ambient concentration and a release rate. For term releases, the user specifies the release duration and the total quantity of material released. Area continuous and area term releases are useful in calculating the effects of a release from pools of spilled volatile liquids. The user must enter the effective radius of the release (i.e., the radius of the circle encompassing the spill area). Also entered is the temperature of the pool and ambient temperature to establish the release rate from a liquid spill. An upwind virtual point source, which results in an initial lateral diffusion equal to the effective radius of the area source, is used to model an area release.

By specifying a release quantity, duration, and area, the user effectively proposes a release rate per unit spill area. EPIcode™ confirms that the volatility of the spilled substance can support such a release rate. If the proposed release rate exceeds the saturation conditions at the release temperature, the EPIcode™ calculates a lower release rate and a corresponding longer release time.

In calculating effective release height, the actual plume height may not be the physical release height (e.g., the stack height). Plume rise can occur because of the velocity of a stack emission and the temperature differential between the stack effluent and the surrounding air. EPIcode™ calculates both the momentum and buoyant plume rise and chooses the greater of the two results.

Concentrations of chemical and radiological materials are highly dependent upon the effective release height (e.g., the effective height of a stack or an evaporating pool of spilled material). Thermal buoyancy was taken into consideration for those scenarios involving fire or heat sources. In those cases, a temperature of 200 °C (392 °F) was assumed for the thermal buoyancy term. This is conservative, since expected surface temperatures and resulting buoyancy terms are expected to be greater in actual fires or heat sources.

In this application, the standard terrain calculation of EPIcode™ is always used. Except as otherwise noted, both the 50th and 95th percentile meteorological (stability class and wind speed) conditions for INEEL were input into EPIcode™. The receptor height is always ground level (0 meters) and the mixing layer height is always 400 meters (1,300 feet).

As described in its user manual (Homann 1988), the EPIcode™ also performs the following steps:

- Treats a release as instantaneous versus continuous, depending upon the plume length at the specific downwind location being considered
- Corrects the concentration for sampling time
- Adjusts the wind speed for release height

- Depletes the plume as a function of downwind distance
- Adjusts the standard deviations of the crosswind and vertical concentrations for brief releases

As output, EPIcode™ can generate data plots of mean toxic chemical concentration (during a specified averaging time) as a function of downwind distance. From these graphs and numerical output, the concentrations at receptor locations are determined and evaluated for health effects.

EPIcode™ was selected as the computer code for release analysis of chemicals amenable to Gaussian modeling after comparison with a number of codes, primarily CHARM and ARCHIE. It was judged easier to use for this simple application than either the more sophisticated, proprietary CHARM code or the comparable, public domain ARCHIE code. The SLAB code had previously been selected by INEEL as the most appropriate of the refined dispersion models (such as CHARM) for modeling special case releases, such as dense gas dispersion, where negative buoyancy effects must be considered. However, because chemical accident scenarios involving dispersion of denser-than-air gases were not considered in this analysis, the SLAB model was not used. EPIcode™ was judged to be a satisfactory code for the inventory of chemicals analyzed.

F.3.1.2 Health Effects

Hazardous constituents dispersed during an accident could induce adverse health effects among exposed individuals. This possible impact is assessed by comparing the airborne concentrations of each substance at specified downwind receptor locations to standard accident exposure guidelines for chemical toxicity.

Where available, the Emergency Response Planning Guideline (ERPG) values were used for this comparison. The guideline values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects. The ERPG values are specific for each substance, and are derived for each of three general severity levels:

- *ERPG-1:* The maximum airborne concentration below which it is believed nearly all individuals could be exposed for up to one hour without experiencing other than mild transient adverse health effects or perceiving a clearly defined objectionable odor.

- *ERPG-2:* The maximum airborne concentration below which it is believed nearly all individuals could be exposed for up to one hour without experiencing or developing irreversible or other serious health effects or symptoms that could impair their abilities to take protective action.

- *ERPG-3:* The maximum airborne concentration below which it is believed nearly all individuals could be exposed for up to one hour without experiencing or developing life-threatening health effects.

Where ERPG values were not derived for a toxic substance, other chemical toxicity values were substituted, as follows:

- For ERPG-1, threshold-limit value/time-weighted average values (ACGIH 1994) were substituted: The time-weighted average is the concentration for a normal 8-hour workday and a 40-hour workweek, to which nearly all workers may be repeatedly exposed, day after day, without adverse effect.

- For ERPG-2, level-of-concern values (equal to 0.1 of immediately-dangerous-to-life-or-health values) were substituted: "level of concern" is defined as the concentration of a hazardous substance in air, above which there could be serious irreversible health effects or death as a result of a single exposure for a relatively short period of time (EPA 1987).

- For ERPG-3, immediately-dangerous-to-life-or-health values were substituted: "immediately dangerous to life or health" is defined as the maximum concentration from which a person could

escape within 30 minutes without a respirator and without experiencing any escape-impairing or irreversible side effects (HHS 1997).

Possible health effects associated with exceeding an ERPG-2 or -3 value are specific for each substance of concern, and must be characterized in that context. When concentrations are found to exceed an ERPG or substitute value, specific toxicological effects for the chemicals of concern are considered in describing possible health effects associated with exceeding a threshold value.

The ERPG values are based upon a one-hour exposure of a member of the general population. In this analysis, the ERPG values were applied only to time-averaged exposures of one hour or less in duration. This approach provides an additional element of conservatism in the evaluation of accidents with releases that are significantly less than one hour. In instances of very short exposures to substances whose effects are concentration-dependent (e.g., chlorine) and where toxicological data support analysis at short exposure times, threshold concentrations of lethality are reported (the minimum concentration necessary to cause a fatality).

F.3.2 Accident Scenario Selection and Descriptions

F.3.2.1 Toxic Chemical Accidents at ANL-W

This section describes the nonradiological consequences of the abnormal event associated with handling uranium ingots. Four accidents have been identified at ANL-W that have the potential to result in the release of either uranium or uranium and cadmium. These accidents, a uranium handling accident, a design-basis uranium fire, a design-basis earthquake, and a beyond-design-basis earthquake, are discussed below.

F.3.2.1.1 Uranium Handling Accident

Uranium ingots (20 percent enrichment or less) from the electrometallurgical treatment process are transferred from the Fuel Conditioning Facility to onsite storage at the Zero Power Physics Reactor Material Building (ANLW-792). Transfers are made using a forklift or by truck. The uranium ingots weigh about 6 kilograms (13 pounds) each. They are stored in containers holding about 140 kilograms (310 pounds) of ingots. Depleted uranium also is stored at ANL-W in containers holding 1,350 kilograms (3,000 pounds) of ingots.

The accident involves a handling accident in which an ingot of uranium is dropped onto a hard surface, small particles are broken off the ingot, and the pyrophoric properties of the uranium result in ignition of the particles. The resulting small fire is assumed to consume 10 percent of the ingot. The accident could occur as a result of a container drop during handling, a drop during inspection, or due to an earthquake. The release occurs at ground level. A handling accident resulting in the drop of a uranium ingot may be anticipated to occur over the life of the project (or about 1 in 10 years). The conditional probability of a fire that consumes 10 percent of the dropped ingot was assumed to be 1 in 10 drops at most. The estimated frequency of the accident is therefore 0.01 per year.

The material at risk is one 6-kilogram ingot of uranium. The damage ratio is 0.1, as it was assumed that 10 percent of the ingot would be consumed in the fire. The airborne release fraction is 0.0001, and the respirable fraction is 1 for metal fires (DOE 1994b). The accident was assumed to occur outdoors or with little confinement. A leak path factor of 1 was assumed. This information is summarized in **Table F–58**.

Table F–58 Toxic Chemical Source Term for a Uranium Handling Accident

Chemical	*Material at Risk (kilograms)*	*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Released (kilograms)*
Uranium	6	0.1	0.0001	1	1	0.00006

F.3.2.1.2 Design-Basis Uranium Fire

Uranium ingots (20 percent enrichment or less) from the electrometallurgical treatment process are transferred from the Fuel Conditioning Facility to onsite storage at the Zero Power Physics Reactor Material Building (ANLW-792). Transfers are made using a forklift or by truck. The uranium ingots weigh about 6 kilograms (13 pounds) each. They are stored in containers holding about 140 kilograms (310 pounds) of ingots. Depleted uranium also is stored at ANL-W in containers holding 1,350 kilograms (3,000 pounds) of ingots.

The accident involves a fire consuming the equivalent of one container of uranium (140 kilograms). The accident could occur due to a handling accident, poor housekeeping in the storage area, electrical failure, or an earthquake. The uranium is in the form of ingots that have a small surface-area-to-mass ratio. Uranium is stored in metal containers that are not combustible. The postulated accident was estimated to have a frequency of 1×10^{-5} per year (see the discussion of radiological accidents in Section F.2).

The material at risk is one 140-kilogram container of uranium. The damage ratio is 1, as it was assumed that all of the uranium would be consumed in the fire. The airborne release fraction is 0.0001, and the respirable fraction is 1 for metal fires (DOE 1994b). The accident was assumed to occur outdoors or with little confinement (e.g., an open storage facility door). A leak path factor of 1 was assumed. This information is summarized in **Table F–59**.

Table F–59 Toxic Chemical Source Term for a Uranium Fire

Chemical	*Material at Risk (kilograms)*	*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Release (kilograms)*
Uranium	140	1	0.0001	1	1	0.014

F.3.2.1.3 Design-Basis Earthquake – Multifacility Effects

This event is the same event as described under radiological accidents for the electrometallurgical treatment of sodium-bonded spent nuclear fuel at ANL-W. The material at risk and release fraction values are summarized in **Table F-60**.

Table F–60 Toxic Chemical Source Term for a Design-Basis Earthquake

Chemical	*Material at Risk (kilograms)*	*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Release (kilograms)*
Uranium	17	1	2.5×10^{-6}	2.5×10^{-6}	1	0.000043

F.3.2.1.4 Beyond-Design-Basis Earthquake – Multifacility Effects

This event is the same event as described under radiological accidents for electrometallurgical treatment at ANL-W. The airborne release fraction and respirable fraction values for cadmium are each 2.5×10^{-6} (Slaughterbeck et al. 1995). The material at risk and release fraction values are summarized in **Table F–61**.

Table F–61 Toxic Chemical Source Term for a Beyond-Design-Basis Earthquake

Chemical	*Material at Risk (kilograms)*	*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Release (kilograms)*
Cadmium	1,000	1	2.5×10^{-6}	2.5×10^{-6}	1	0.0025
Uranium	17	1	2.5×10^{-6}	2.5×10^{-6}	1	0.000043

F.3.2.1.5 Liquid Sodium Fire

This event is the same event as described under radiological accidents for melt and dilute processing at ANL-W. The accident is associated with the fuel cleaning process used during the melt and dilute process or in preparation of the fuel for shipment to SRS for processing.

The accident involves a fire during the declad and clean processing of the spent nuclear fuel due to a breach of the Hot Fuel Examination Facility and exposure of liquid sodium to the air. The most probable cause of air in-leakage is expected to be an earthquake. As discussed in the radiological accident description, this event was assumed to occur with a frequency of 0.008 per year. The material at risk would be the sodium cleaned from the spent nuclear fuel and was conservatively estimated to be half of all of the sodium contained in the spent nuclear fuel, 300 kilograms. The release fraction values are provided in **Table F–62**. The assumption that all of the sodium would be converted to sodium hydroxide and volatilized by the fire results in the airborne release fraction and respirable fraction values of 1 each.

Table F–62 Toxic Chemical Source Term for a Sodium Fire in the Hot Fuel Examination Facility

Chemical	*Material at Risk (kilograms)*	*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Release (kilograms)*
Sodium	330	1	1	1	0.125	41.3

F.3.2.2 Toxic Chemical Accidents at SRS

The SRS Spent Nuclear Fuel Management Final EIS (DOE 2000) analyzed the consequences of accidental releases of hazardous chemicals for operations located in F-Area. These accidents involve the spill of materials associated with the wet storage of spent nuclear fuel in F-Area. These are generic-type accidents that are independent of processing cleaned and declad blanket fuel pins at either F-Canyon or Building 105-L. The activities associated with processing the cleaned and declad blanket spent nuclear fuel are not expected to result in the introduction of additional hazardous materials or additional accident scenarios. Therefore, the accident scenarios identified in the SRS Spent Nuclear Fuel Management Draft EIS were selected to represent the hazardous chemical accidents associated with processing sodium-bonded spent nuclear fuel.

F.3.3 Accident Analyses Consequences and Risk Results

Tables F–63 through **F–67** provide the chemical risk calculation results for electrometallurgical treatment process-related accidents at the ANL-W facility. **Table F–68** reproduces the consequences from hazardous chemical accidents at SRS, as originally developed for the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000).

Table F–63 Summary of Toxic Chemical Exposure Results for a Uranium Handling Accident at ANL-W

Receptor Location	*Chemical*	*Concentration (milligrams per cubic meter)*	*Fraction of ERPG-1*	*ERPG-1 Value*
Noninvolved worker	Uranium	0.000177	0.000295	0.6 mg/m^3
Maximally exposed offsite individual	Uranium	1.14×10^{-8}	1.9×10^{-8}	0.6 mg/m^3

mg/m^3 = milligrams per cubic meter.

Table F–64 Summary of Toxic Chemical Exposure Results for a Uranium Fire at ANL-W

Receptor Location	*Chemical*	*Concentration (milligrams per cubic meter)*	*Fraction of ERPG-1*	*ERPG-1 Value*
Noninvolved worker	Uranium	0.0413	0.0688	0.6 mg/m^3
Maximally exposed offsite individual	Uranium	2.7×10^{-6}	4.4×10^{-6}	0.6 mg/m^3

mg/m^3 = milligrams per cubic meter.

Table F–65 Summary of Toxic Chemical Exposure Results for a Design-Basis Earthquake at ANL-W

Receptor Location	*Chemical*	*Concentration (milligrams per cubic meter)*	*Fraction of ERPG-1*	*ERPG-1 Value*
Noninvolved worker	Uranium	100 meters: 1.29×10^{-7} 230 meters: 1.03×10^{-6}	100 meters: 2.15×10^{-7} 230 meters: 1.72×10^{-6}	0.6mg/m^3
Maximally exposed offsite individual	Uranium	5.25×10^{-8}	8.75×10^{-8}	0.6 mg/m^3

mg/m^3 = milligrams per cubic meter.

Table F–66 Summary of Toxic Chemical Exposure Results for a Beyond-Design-Basis Earthquake at ANL-W

Receptor Location	*Chemical*	*Concentration (milligrams per cubic meter)*	*Fraction of ERPG-1*	*ERPG-1 Value*
Noninvolved worker	Cadmium	7.5×10^{-6}	0.00025	0.03 mg/m^3
	Uranium	1.27×10^{-7}	2.12×10^{-7}	0.6 mg/m^3
Maximally exposed offsite individual	Cadmium	3.10×10^{-6}	0.0001	0.03 mg/m^3
	Uranium	5.3×10^{-8}	8.8×10^{-8}	0.6 mg/m^3

mg/m^3 = milligrams per cubic meter.

Table F–67 Summary of Toxic Chemical Exposure Results for a Sodium Fire at ANL-W

Receptor Location	*Chemical*	*Concentration (milligrams per cubic meter)*	*Fraction of PEL-TWA*	*PEL-TWA*
Noninvolved worker	Sodium hydroxide	0.15	0.075	2 mg/m^3
Maximally exposed offsite individual	Sodium hydroxide	0.002	0.001	2 mg/m^3

PEL-TWA = Permissible Exposure Limits–Time-Weighted Average, mg/m^3 = milligrams per cubic meter.
[a] No ERPG value is available for sodium hydroxide; therefore, PEL-TWA was used instead.

Table F–68 Summary of Toxic Chemical Exposure Results for a Wet Storage Container Rupture at SRS

Frequency (event/year)	*Receptor*	*Chemical*	*Concentration* [a]	*Fraction of PEL-TWA*	*PEL-TWA*
0.005	Noninvolved worker	Sodium hydroxide	less than PEL-TWA	N/A[b]	2 mg/m^3
0.005	Noninvolved worker at 640 meters	Nitric acid	3.1×10^{-3} mg/m^3	0.00062	5 mg/m^3
	Maximally exposed offsite individual		4.0×10^{-4} mg/m^3	0.00008	5 mg/m^3
0.005	Noninvolved worker	Sodium nitrite	6.0×10^{-3} mg/m^3	0.0012 [2]	2 mg/m^3 [c]

PEL-TWA = Permissible Exposure Limits–Time-Weighted Average, mg/m^3 = milligrams per cubic meter.
[a] SRS Spent Nuclear Fuel Management Final EIS (DOE 2000).
[b] Not available – SRS Spent Nuclear Fuel EIS states that concentration only in less than the lowest PEL-TWA.
[c] No PEL-TWA for this specific chemical. Lowest PEL-TWA of potential chemical reaction products is 2 milligrams per cubic meter.

Table F-69 provides a summary of the applicability of the analyzed toxic chemical accidents to each of the alternatives considered in detail for processing the sodium-bonded spent nuclear fuel. The hazardous chemical accidents applicable to the No Action Alternative include only those accidents associated with operation at ANL-W. Additionally, only three of the four accidents identified, excluding the beyond-design-basis earthquake, can be associated with this alternative. Accidents associated with this alternative are the result of activities from the final processing of the sodium-bonded spent nuclear fuel treated with the electrometallurgical treatment process as part of the Electrometallurgical Treatment Demonstration Program. Alternatives 2 through 5 include electrometallurgical treatment of at least some of the sodium-bonded spent nuclear fuel and decladding and cleaning of blanket spent nuclear fuel; therefore, all of the identified toxic chemical accidents at ANL-W are applicable to these alternatives. Alternative 1 includes electrometallurgical treatment of fuel, but no decladding and cleaning operations; therefore, for this alternative, all ANL-W accidents except the sodium fire are applicable. Processing of the spent nuclear fuel at SRS occurs only in Alternatives 3 and 5, and the accidents at SRS are applicable to these two alternatives. The accidents identified for ANL-W are associated with the electrometallurgical treatment of the sodium-bonded spent nuclear fuel. Alternative 6 does not include this treatment option and no other accidental releases of hazardous chemicals were identified.

Table F–69 Applicability of Hazardous (Toxic) Chemical Accidents to Sodium-Bonded Spent Nuclear Fuel Alternatives

	Alternative	*ANL-W Toxic Chemical Accidents*	*SRS Toxic Chemical Accidents*
	No action	Uranium handling accident Uranium fire Design-basis earthquake	Not applicable
1	Electrometallurgically treat blanket and driver fuel at ANL-W	Uranium handling accident Uranium fire Design-basis earthquake Beyond-design-basis earthquake	Not applicable
2	Clean and package blanket fuel in high-integrity cans and electrometallurgically treat driver fuel at ANL-W	Alternative 1 accidents plus sodium fire	Not applicable
3	Declad and clean blanket fuel and electrometallurgically treat driver fuel at ANL-W; PUREX process blanket fuel at SRS	Alternative 1 accidents plus sodium fire	Wet storage, container rupture
4	Melt and dilute blanket fuel and electrometallurgically treat driver fuel at ANL-W	Alternative 1 accidents plus sodium fire	Not applicable
5	Declad and clean blanket fuel and electrometallurgically treat driver fuel at ANL-W; melt and dilute blanket fuel at SRS	Alternative 1 accidents plus sodium fire	Wet storage, container rupture
6	Melt and dilute blanket and driver fuel at ANL-W	Sodium fire	Not applicable

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Appendix G

Evaluation of Human Health Effects from Overland Transportation

APPENDIX G
EVALUATION OF HUMAN HEALTH EFFECTS FROM OVERLAND TRANSPORTATION

G.1 INTRODUCTION

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. 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 spent nuclear fuel are assessed.

This appendix provides an overview of the approach used to assess the human health risks that may result from overland transportation. The topics in this appendix include the scope of the assessment, packaging and determination of potential transportation routes, analytical methods used for the risk assessment (e.g., computer models), and important assessment assumptions. It also 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 risk assessment results are presented in this appendix in terms of "per-shipment" risk factors, as well as for the total risks for a given alternative. Per-shipment risk factors provide an estimate of the risk from a single shipment. The total risks for a given alternative are found by multiplying the expected number of shipments by the appropriate per-shipment risk factors.

G.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 environmental impact statement (EIS) estimates the human health risks associated with the transportation of sodium-bonded spent nuclear fuel for all alternatives. There are several shipping arrangements for various fuel types that cover all alternatives evaluated. Consistent with the scope of the overland transportation human health risks, this evaluation focuses on using onsite and offsite public highways.

Transportation-Related Activities

The transportation risk assessment is limited to estimating the human health risks incurred during overland transportation for each alternative. The risks to workers or to the public during loading, unloading, and handling prior to or after shipment are not included in the overland transportation assessment, but are addressed in Appendix F of this EIS. 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.

All radiological 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 (see 10 CFR 20), which is the sum of the effective dose equivalent from external radiation exposure and the 50-year committed effective dose equivalent from internal radiation exposure. Radiation doses are presented in units of roentgen equivalent man (rem) for individuals and person-rem for collective populations. The impacts are further expressed as health risks in terms of latent cancer fatalities and cancer incidence in exposed populations using the dose-to-risk conversion factors established by the National Council on Radiation Protection and Measurement (NCRP 1993).

Nonradiological Impacts

In addition to the radiological risks posed by overland transportation activities, vehicle-related risks are also assessed for nonradiological causes (i.e., causes related to the transport vehicles and not the radioactive cargo) for the same transportation routes. The nonradiological transportation risks, which would be incurred for similar shipments of any commodity, are assessed for both incident-free and accident conditions. The 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 of cargo. State-specific transportation fatality rates are used in the assessment. Nonradiological risks are presented in terms of estimated fatalities.

Transportation Modes

All shipments are assumed to take place by truck transportation modes.

Receptors

Transportation-related risks are calculated and presented separately for workers and members of the general public. The workers considered are truck crew members involved in the actual overland transportation. The general public includes all persons who could be exposed to a shipment while it is moving or stopped during transit. The affected population includes individuals living within 800 meters (0.5 miles) of each side of the road. Potential risks are estimated for the affected populations and for the hypothetical maximally exposed individual. For incident-free operation, the maximally exposed individual would be an individual stuck in traffic next to the shipment for 30 minutes. For accident conditions, the maximally exposed individual would be an individual located 33 meters (108 feet) directly downwind from the accident. The risk to the affected population is a measure of the radiological risk posed to society as a whole by the alternative being considered. As such, the impact to the affected population is used as the primary means of comparing various alternatives.

G.3 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 promote safety is 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 transportation risk assessment. Regulatory packaging requirements are discussed briefly below. The representative packaging and shipment configurations assumed for this EIS also are described below.

G.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 as specified in 49 CFR 172 and 173 and 10 CFR 71.

Transportation packaging for small quantities of radioactive materials must be designed, constructed, and maintained to contain and shield their contents during normal transport conditions. For large quantities and for more highly radioactive material, such as high-level radioactive waste or 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. Four basic types of packaging are used: Excepted, Industrial, Type A, and Type B. Another packaging option, "Strong, Tight," is still available for some domestic shipments.

Excepted packages are limited to transporting materials with extremely low-levels of radioactivity. Industrial packages are used to transport materials that, because of their low concentration of radioactive materials, present a limited hazard to the public and the environment. Type A packages are designed to protect and retain their contents under normal transport conditions and must maintain sufficient shielding to limit radiation exposure to handling personnel. These packages are used to transport radioactive materials with higher concentrations or amounts of radioactivity than Excepted, or Industrial packages. Strong, Tight packages are used in the United States for shipment of certain materials with low-levels of radioactivity, such as natural uranium and rubble from the decommissioning of nuclear reactors. Type B packages are used to transport material with the highest radioactivity levels, are designed to protect and retain their contents under transportation accident conditions, and are described in more detail in the following sections.

G.3.2 Regulations Applicable to Type B Casks

Regulations for the transport of radioactive materials in the United States are issued by the U.S. Department of Transportation and are codified in 49 CFR 173. The regulation authority for radioactive materials transport is jointly shared by the U.S. Department of Transportation and the NRC. As outlined in a 1979 Memorandum of Understanding with the NRC, the U.S. 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 U.S. Department of Transportation also regulates the labeling, classification, and marking of all spent nuclear fuel packages. The 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.

Department of Energy (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 173, "Shippers--General Requirements for Shipping and Packaging." DOE Headquarters can issue a certificate of compliance for a package to be used only by DOE and its contractors.

G.3.2.1 Cask Design Regulations

Spent nuclear fuel is transported in robust "Type B" transportation casks that are certified for transporting radioactive materials. 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.

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 (described in the next section) for the cask and that the cask will function as designed.

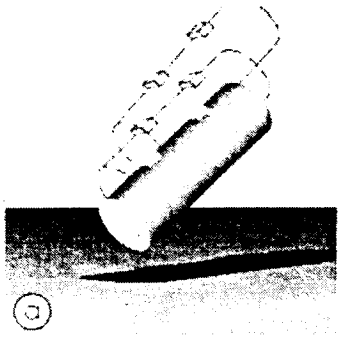
Regardless of where a transportation cask is designed, fabricated, or certified for use, it must meet certain minimum performance requirements (10 CFR 71.71–71.77). The primary function of a transportation cask is to provide containment and shielding. Regulations require that casks must be operated, inspected, and maintained to high standards to ensure 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. Further, a number of obsolete casks have been tested under severe accident conditions to demonstrate their adherence to design criteria without failure.

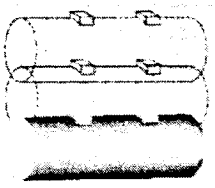
Transportation casks are built out of heavy, durable structural materials such as stainless steel. These materials must ensure cask performance under a wide range of temperatures (10 CFR 71.43). In addition to the structural materials, shielding is provided to limit radiation levels at the surface and 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. The assemblies are supported by internal structures, called baskets, that provide shock and vibration resistance and establish minimum spacing and heat transfer to maintain the temperature of the contents within the limits specified in the Safety Analysis Report for Packaging.

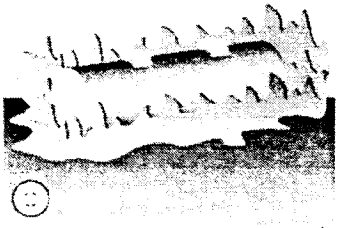
Finally, to limit impact forces and minimize damage to the structural components of a cask in the event of a transportation accident, impact-absorbing structures may be attached to the exterior of the cask. These are usually composed of balsa wood, foam, or aluminum honeycomb that is designed to readily 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.

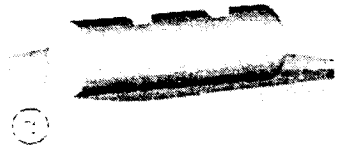
G.3.2.2 Design Certification

For certification, transportation casks must be shown by analysis and/or testing 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 G–1**. The NRC issues regulations (10 CFR 71) governing the transportation of radioactive materials. In addition to the tests shown in Figure G–1, the regulations affecting Type B casks require that a transportation cask with activity greater than 10^6 curies (which is applicable to spent nuclear fuel) be designed and constructed so that its undamaged containment system would withstand









Standards for Type B Casks

For certification by the U.S. Nuclear Regulatory Commission, a cask must be shown by test or analysis to withstand a series of accident conditions without releasing its contents. These conditions have been internationally accepted as simulating damage to spent nuclear 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. An undamaged containment system is subjected to a deep water-immersion test. The details of the tests are as follows:

Impact

Free Drop (a) – The cask drops 9 meters (30 feet) onto a flat, horizontal, unyielding surface so that it strikes at its weakest point.

Puncture (b) – The cask drops 1 meter (40 inches) onto a 15.2-centimeter (6-inch) diameter steel bar at least 20.3 centimeters (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 802 °C (1,475 °F) thermal environment for 30 minutes.

Water Immersion (d)

The cask is completely submerged under at least 1 meter (40 inches) of water for 8 hours. Additionally, undamaged containment systems (casks) are required to withstand more rigorous immersion tests.

Figure G–1 Standards for Transportation Casks

an external water pressure of 2 megapascals (290 pounds per square inch), or immersion in 200 meters (656 feet) of water, for a period of not less than one hour without collapse, buckling, or allowing water to leak into the cask.

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, maintenance program, and the quality assurance program to be used for design and fabrication. Upon completion of a satisfactory review of the Safety Analysis Report for Packaging to verify compliance to the regulations, a Certificate of Compliance is issued.

G.3.2.3 Transportation Regulations

To ensure 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 before shipping radioactive material. 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. DOE will be the Shipper-of-Record for the shipments that could be sent.

External radiation from a package must be below specified limits that minimize the exposure of handling personnel and the general public. For these types of shipments, the external radiation dose rate during normal transportation conditions must be maintained below the following limits of 49 CFR 173:

- 10 millirem per hour at any point 2 meters (6.6 feet) 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 millirem per hour in any normally occupied position in the transport vehicle

Additional restrictions apply to package surface contamination levels, but these restrictions are not important for the transportation radiological risk assessment. Current contamination standards assure that workers and public receive doses much lower than those associated with radiation emitted from the casks. 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 different Type B designs would be expected to perform similarly during incident-free and accident transportation conditions. The specific containers selected or designed, however, will determine the total number of shipments necessary to transport a given quantity material.

G.3.2.4 Communications

Proper communication assists in ensuring safe preparation and handling of transportation casks. Communication is provided by labels, markings, placarding, 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 by giving the transport index. The transport index lists the ionizing radiation level (in millirem per hour) at a distance of 1 meter (3.3 feet) from the cask surface.

In addition to the label requirements, markings (49 CFR 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 (49 CFR 172.203). Transportation casks are required to be permanently marked with the designation "Type B," the owner's (or fabricators') 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. Spent nuclear fuel, which constitutes a highway route-controlled quantity or "HRCQ," 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 for the spent nuclear fuel should contain the notation "HRCQ" and have entries identifying the following: the name of the shipper, emergency response telephone number, description of contents, and the shipper's certificate, as described in 49 CFR 172 Subpart C.

In addition, drivers of motor vehicles transporting radioactive material must have been trained in accordance with the requirements of 49 CFR 172.700. The training requirements include familiarization with the regulations, emergency response information, and the communication programs required by the Occupational Safety and Health Administration. Drivers are also required to have been trained on the procedures necessary for safe operation of the vehicle used to transport the spent nuclear fuel.

G.3.3 Packages Used in the Transportation of Spent Nuclear Fuel

Two Type B casks, a formerly certified Type B cask, and an NRC-certified cask would provide primary transportation services for sodium-bonded fuel where public roads are involved. A commercially available cask would be certified and used for single shipments of miscellaneous sodium-bonded fuel from Tennessee and New Mexico. One other cask for onsite fuel transfers at ANL-W which does not use public roads will be employed. It is discussed below.

The TN-FSV is a certified Type B cask that would be used for intrasite transportation, and NAC-LWT would be used for the intersite transportation. The Peach Bottom (PB-1) is a formerly certified Type B cask that would be used for some of the intrasite transportation. The NRC-certified T-3 cask would be used for shipping the Fast Flux Test Facility Driver fuel from Washington to Idaho. The NRC-license is equivalent to the Type B certification described in the earlier sections.

The TN-FSV cask is a steel and lead shielded shipping cask originally designed for high temperature gas–cooled reactor fuel elements from the Fort St. Vrain reactor. The cask is a right circular cylinder, with a balsa and redwood impact limiter at each end. The cask body is made of two concentric shells of type 304 stainless steel, welded to a bottom plate and a top closure flange. The inner shell has an inside diameter of 46 centimeters (18 inches) and is 2.8 centimeters (1.1 inches) thick, and the cavity is 505 centimeters (199 inches) long. The outer shell has an outside diameter of approximately 76 centimeters (30 inches) and is 3.8 centimeters (1.5 inches) thick. The gross package weight, including the contents, is 21,319 kilograms (47,000 pounds). **Figure G–2** shows the TN-FSV.

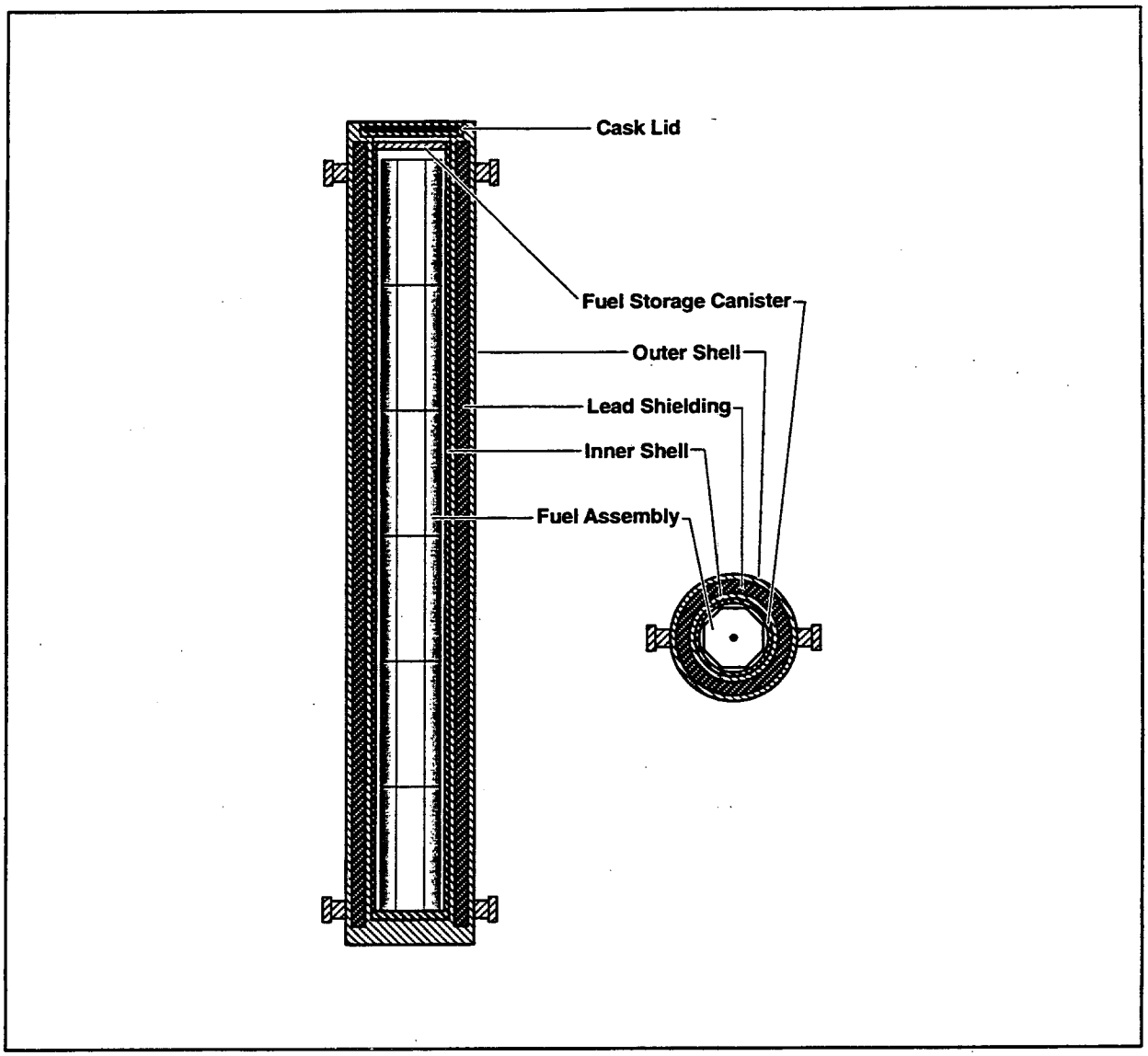


Figure G–2 TN-FSV Cask

The TN-FSV cask first received an NRC Certificate of Compliance in March 1993, and this certificate has been supplemented several times since that time. The current Certificate of Compliance expires in May 2004. The Certificate of Compliance would have to be supplemented for the materials that could be carried in this program. In addition to the size of the cavity, the limiting factors for this cask on the current Certificate of Compliance are a maximum of 360 watts of decay heat and a maximum total weight of contents of 2268 kilograms (5,000 pounds), including the fuel elements, fuel storage container and shield plug (NRC 1998).

The NAC-LWT is a steel encased lead shielded shipping cask. The overall dimensions with impact limiters are 589 centimeters (232 inches) long by 165 centimeters (65 inches) in diameter. The cask body is approximately 508 centimeters (200 inches) in length and 112 centimeters (44 inches) in diameter. The cask cavity is approximately 0.41 cubic meters (14.5 cubic feet). The maximum weight of the package is 23,587 kilograms (52,000 pounds) and the maximum weight of the contents and basket is 1,814 kilograms (4,000 pounds). **Figure G–3** shows the NAC-LWT.

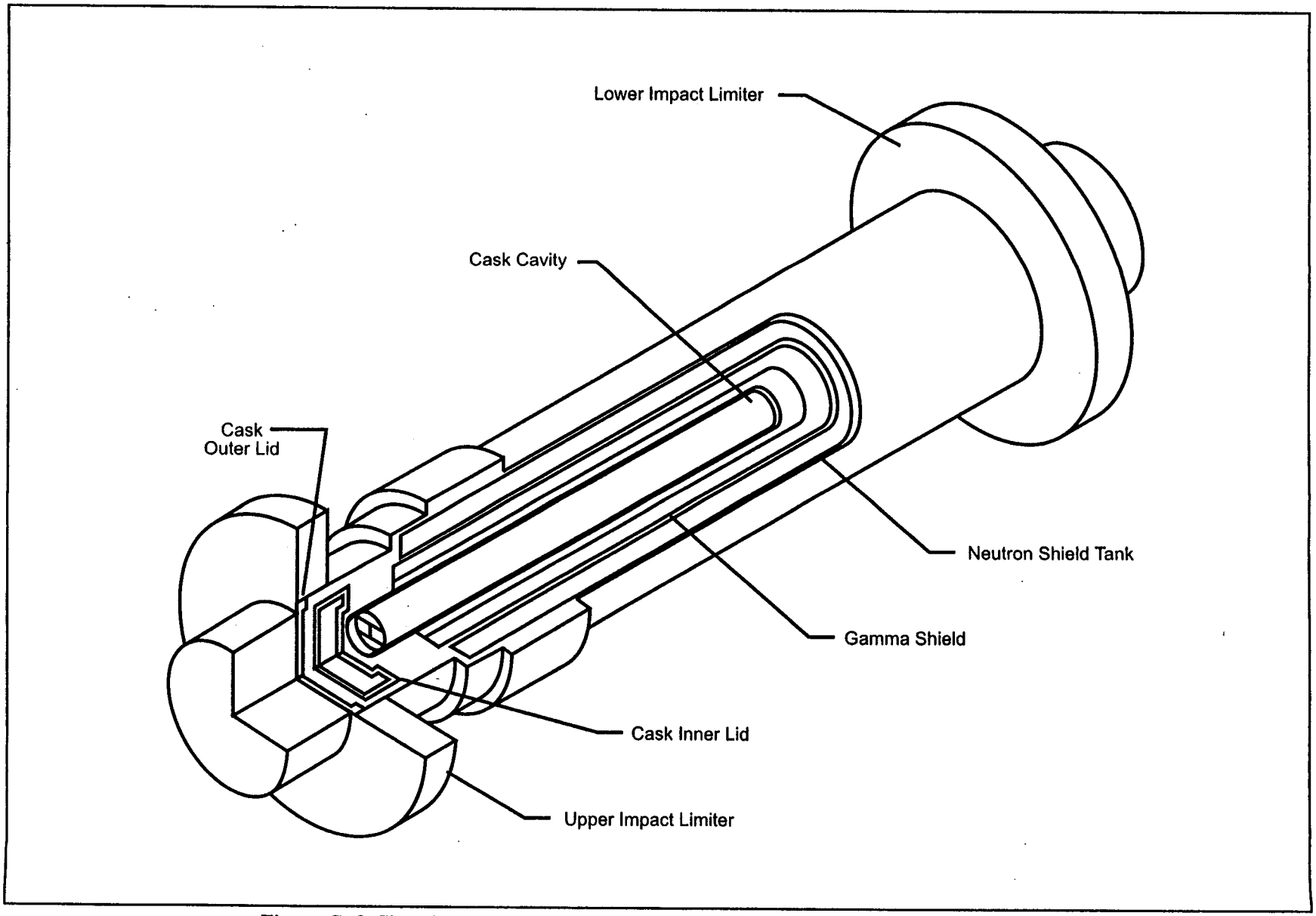


Figure G–3 Simplified Drawing of a NAC-LWT (Legal Weight Truck) Shipping Cask

The NAC-LWT first received an NRC Certificate of Compliance in March 1995, and this certificate has been supplemented several times. The current Certificate of Compliance expired in February 2000; it is likely that it will be renewed. The Certificate of Compliance would not need to be supplemented for the materials that could be carried in this program. The cask is designed to carry up to 42 reactor fuel assemblies. Besides the size of the cavity and weight, the limiting factor for this cask on the current Certificate of Compliance is a maximum of 210 watts of decay heat.

The intrasite transportation of Fermi-1 blanket fuel would use the formerly certified PB-1 cask. This cask was originally licensed for carrying Peach Bottom fuel, and was used to bring the Fermi-1 spent nuclear fuel to the Idaho Nuclear Technology and Engineering Center (INTEC). The Certificate of Compliance for this cask has expired. Since the movement is a short distance on closed DOE-controlled roads, DOE procedures and NRC regulations do not require the use of a certified Type B cask. The use of formerly certified casks provides a margin of safety beyond that required by NRC regulations. The level of safety for intrasite shipments is carefully controlled by internal procedures, and the level of protection given by the PB-1 cask is approximately equivalent to that of a certified Type B cask. Since the roads are closed and site is uninhabited, there would be no measurable impact to the public.

The Experimental Breeder Reactor II (EBR-II) driver and blanket fuel currently in storage at Argonne National Laboratory-West (ANL-W) is stored in HFEF-5 sealed canisters. The canisters are single use, welded steel cans. DOE packs these cans in an unlicensed HFEF-5 cask for onsite shipping. Fast Flux Test Facility driver material currently in storage at the Hanford Site would be shipped in the NRC-certified T-3 cask.

Waste from ANL-W will be shipped to the Idaho National Engineering and Environmental Laboratory (INEEL) Dry Transfer Facility in cans designed to closely fit the DOE standardized canisters. Waste includes ceramic waste form, metallic waste form, spent nuclear fuel and melt and dilute product. The standardized canisters are either a 46 centimeter (18-inch) outside diameter with a 0.95 centimeter (0.375 inch) thick pipe or 61 centimeter (24 inch) outside diameter with a 1.27 centimeter (0.5 inch) thick pipe made of Type 316L stainless steel with welded flanges on each end. DOE has not determined which Type B cask will be used to carry these canisters.

G.3.4 Ground Transportation Route Selection Process

According to DOE guidelines, spent nuclear fuel shipments must comply with both the NRC and U.S. Department of Transportation regulatory requirements. NRC regulations cover the packaging and transport of spent nuclear fuel, whereas the U.S. Department of Transportation specifically regulates the carriers and the conditions of transport, such as routing, handling and storage, and vehicle and driver requirements. The highway routing of nuclear material is systematically determined according to U.S. Department of Transportation regulations 49 CFR 171–179 and 49 CFR 397 for commercial shipments. Specific routes cannot be publicly identified in advance for DOE's Transportation Safeguards Division's shipments because they are classified to protect national security interests.

The U.S. Department of Transportation routing regulations require that shipment of a highway route-controlled quantity of radioactive material be transported over a preferred highway network, including interstate highways, with preference toward interstate system bypasses and beltways around cities and state-designated preferred routes. A state or tribe may designate a preferred route to replace or supplement the interstate highway system in accordance with U.S. Department of Transportation guidelines (DOT 1992).

Carriers of highway route-controlled quantities are required to use the preferred network unless they are moving from their origin to the nearest interstate highway or from the interstate highway to their destination, they are making necessary repair or rest stops, or emergency conditions render the interstate highway unsafe or impassable. The primary criterion for selecting the preferred route for a shipment is travel time. Preferred

routing takes into consideration accident rate, transit time, population density, activities, time of day, and day of the week.

The HIGHWAY computer code (Johnson et al. 1993) is used for selecting highway routes in the United States. The HIGHWAY database is a computerized road atlas that currently describes over 386,000 kilometers (240,000 miles) of roads. The Interstate System and all U.S. (US-designated) highways are completely described in the database. In addition, most of the principal state highways and many local and community roads 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. Features in the HIGHWAY code allow the user to select routes that conform to U.S. Department of Transportation regulations. Additionally, the HIGHWAY code contains data on the population densities along the routes. The distances and populations from the HIGHWAY code are part of the information used for the transportation impact analysis in this EIS.

G.4 Methods for Calculating Transportation Risks

The overland transportation risk assessment method is summarized in **Figure G–4**. After the EIS alternatives were identified and the goals of the shipping campaign were understood, data was collected on material characteristics and accident parameters. Accident parameters were largely based on the NRC studies of transportation accidents undertaken for the *Final Environmental Impact Statement on the Transportation of Radioactive Material by Air and Other Modes* (NRC 1977) and the Modal Study (NRC 1987).

Representative routes that may be used for the shipments were selected for risk assessment purposes using the HIGHWAY code. They do not necessarily represent the actual routes that would be used to transport nuclear materials. Specific routes cannot be identified in advance because the routes cannot be finalized until they have been reviewed and approved by the NRC. The selection of the actual route would be responsive to environmental and other conditions that would be in effect or could be predicted at the time of shipment. Such conditions could include adverse weather conditions, road conditions, bridge closures, and local traffic problems. For security reasons, details about a route would not be publicized before the shipment.

The first analytic step in the ground transportation analysis was to determine the incident-free and accident risk factors on a per-shipment basis. Risk factors, as with 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 shipping container (cask) and public exposure to the chemical toxicity of the transportation vehicle exhaust. The probability of incident-free exposure is unity (one).

For each alternative, risks were assessed for both incident-free transportation and accident conditions. For the incident-free assessment, risks are calculated for both collective populations of potentially exposed individuals and for maximally exposed individuals. 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 postulated transportation accidents.

The RADTRAN 5 computer code (Neuhauser and Kanipe 1998) is used for incident-free and accident risk assessments to estimate the impacts on population. RADTRAN 5 was developed by Sandia National Laboratories to calculate population risks associated with the transportation of radioactive materials by a

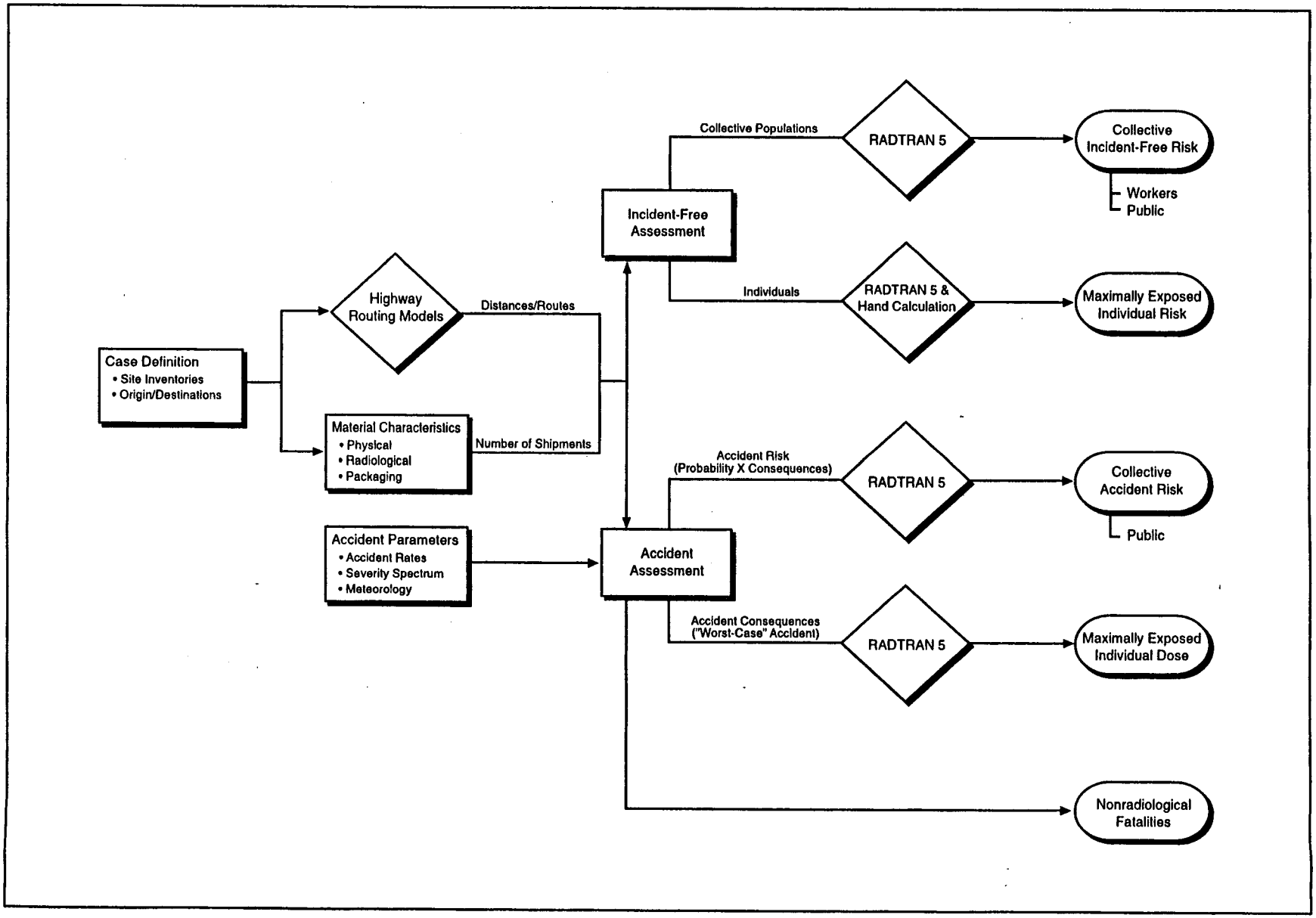


Figure G–4 Overland Transportation Risk Assessment

variety of modes, including truck, rail, air, ship, and barge. RADTRAN 5 was used to calculate the doses to the maximally exposed individuals.

The RADTRAN 5 population risk calculations include both the consequences and probabilities of potential exposure events. The RADTRAN 5 code consequence analyses include the cloud shine, ground shine, inhalation, and resuspension exposures. 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.

G.5 ALTERNATIVES, PARAMETERS, AND ASSUMPTIONS

G.5.1 Material Inventory and Shipping Campaigns

Table G–1 lists the fuel that could be shipped as a result of implementing an alternative to treat sodium-bonded spent nuclear fuel.

Table G–1 Transportation Summary for Sodium-Bonded Fuel

Fuel Type	*Applicable Alternatives* [a]	*Metric Tons of Heavy Metal*	*Origin/State*	*Destination/ State*	*Cask*	*Number of Shipments/Type of Transport*
EBR-II driver	All	1.1	ANL-W/ID	ANL-W/ID	HFEF-5	84/On site, intrafacility transfers
EBR-II driver	All	2.0	INTEC/ID	ANL-W/ID	TN-FSV, or NAC-LWT	17/On site with roads open, or 43/On site with roads open
EBR-II blanket	All	22.4	ANL-W/ID	ANL-W/ID	HFEF-5	165/On site, intrafacility transfers
Fast Flux Test Facility driver [b]	All	0.33	Hanford/WA	ANL-W/ID	T-3	10/Public highways
Fermi-1 blanket	All	34.2	INTEC/ID	ANL-W/ID	PB-1	14/On site with road closed
Miscellaneous [b]	All	0.1	Oak Ridge National Laboratory/TN Sandia National Laboratories/ NM SRS/SC	ANL-W/ID	To be determined by DOE	1/Public highways 1/Public highways 1/Public highways
Declad EBR-II blanket.	3 and 5	22.4	ANL-W/ID	SRS/SC	NAC-LWT	11/Public highways
Declad Fermi-1 blanket	3 and 5	34.2	ANL-W/ID	SRS/SC	NAC-LWT	18/Public highways

[a] "All" includes the proposed action plus the No Action Alternative.
[b] This fuel is assumed to be in Idaho per amended Record of Decision for the Programmatic Spent Nuclear Fuel EIS (61 FR 9441).
Key: ID = Idaho; NM = New Mexico; SC = South Carolina; TN = Tennessee; WA = Washington.

The following shipment campaigns related to sodium-bonded spent nuclear fuel were analyzed by DOE in other National Environmental Policy Act documents and are not treated in detail here.

- Fast Flux Test Facility driver material is currently stored at the Hanford Site, and the transportation impacts are included in the *Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final EIS* (Programmatic Spent Nuclear Fuel EIS) (DOE 1995), and finalized in the Amendment to the Record of Decision (61 FR 9441).

- Miscellaneous spent nuclear fuel is currently stored at the Oak Ridge National Laboratory and at Sandia National Laboratory/New Mexico, and the transportation impacts are included in the Programmatic Spent Nuclear Fuel EIS (DOE 1995), and finalized in the Amendment to the Record of Decision (61 FR 9441).

Impacts of transporting sodium-bonded spent nuclear fuel to INEEL were calculated using a similar approach to that used in the Programmatic Spent Nuclear Fuel EIS (DOE 1995). In the Programmatic Spent Nuclear Fuel EIS, the representative transportation routes were analyzed using HIGHWAY Code (Johnson et al. 1993), and the risks were quantified using RADTRAN 4 Code, an older version of the code used in this EIS. The impact analysis in the Programmatic EIS was based on regulatory limit for cask dose rate and representative fuel isotope inventories. The isotopic inventories of the various sodium-bonded spent nuclear fuel presented in Appendix D are orders of magnitude less than those used in the Programmatic EIS. In addition, shipping cask dose rate containing sodium-bonded spent nuclear fuel would be between two to four orders of magnitude less than the regulatory limit dose rate (SAIC 1999). Therefore, the transportation impacts as presented in the Programmatic Spent Nuclear Fuel EIS would be very conservative for this EIS.

All EBR-II blanket and some EBR-II driver fuel are currently stored at ANL-W and would be subject to a building-to-building movement for processing. Since the movement is a short distance, on closed DOE-controlled roads, DOE procedures and NRC regulations do not require the use of a certified Type B cask. DOE would use the HFEF-5 canister which is the sealed canister in which the spent nuclear fuel is currently stored. No incident-free risk analysis is necessary, because the public would receive no measurable exposure. Worker dose is included in the process and handling dose estimates because the same personnel would be moving the spent nuclear fuel. No accident analysis is necessary because potential accidents during movement are bounded in frequency and consequence by handling accidents. Once the cask is closed for the low-speed movement to the nearby building, the likelihood and consequence of any foreseeable accident are very small and not further quantified.

Fermi-1 blanket fuel would be shipped from the INTEC to ANL-W in the formerly certified Type B cask, the PB-1 Cask. Since DOE would close the roads between INTEC and ANL-W using existing traffic gates, and there are no homes in the vicinity of the road within the INEEL site boundary, no quantitative analysis is necessary. No incident-free risk analysis is necessary, because the public would receive no measurable exposure. Worker dose is included in the process and handling dose estimates because the same personnel would be moving the spent nuclear fuel. Once the cask is closed for the movement on the INEEL site roads, the likelihood and consequence of any foreseeable accident are very small.

EBR-II driver fuel currently stored at INTEC would be shipped to ANL-W in a certified Type B cask, either TN-FSV or NAC-LWT. Since the cask would be certified, DOE would not close the roads between INTEC and ANL-W. However, since there are no homes in the vicinity of the road within the INEEL site boundary, limited quantitative analysis is necessary. No incident-free risk analysis for exposure to the public at stops or in their homes is necessary. Worker dose is analyzed for the transportation crew, and the dose to other vehicles using the road is estimated. No accident analysis is necessary, because potential accidents during movement are bounded in frequency and consequence, by the handling accidents. Once the cask is closed for the movement on the INEEL site roads, the likelihood and consequence of any foreseeable accident are very small and not further quantified.

Waste production canisters generated at ANL-W will be shipped to the INEEL Dry Transfer Facility for eventual shipment to and disposal in a geological repository. These canisters would be shipped in a certified cask, so DOE would not close the roads between INEEL and ANL-W. However, since there are no homes in the vicinity of the road with the INEEL site boundary, limited quantitative analysis is necessary. No incident-free risk analysis for exposure to the public at stops or in their homes is necessary. Worker dose is analyzed for the transportation crew, and the dose to other vehicles using the road is estimated. No detailed accident analysis is necessary because potential accidents during movement are bounded in frequency and consequence by the handling accidents. Once the cask is closed for the movement on the INEEL site roads, the likelihood and consequence of any foreseeable accidents are very small and not further quantified.

EBR-II and Fermi-1 blanket fuel that is cleaned and declad at ANL-W (Alternatives 3 and 5) would be transported to Savannah River Site (SRS) in NAC-LWT casks. The impacts associated with these shipments are analyzed in detail, including incident-free exposure to the truck crew and public, and accident risk. **Table G-2** summarizes the shipping campaigns necessary to complete each alternative.

Table G–2 Summary of Shipments Under Each Alternative

	Spent Nuclear Fuel for Processing				*Waste Production Canisters to INEEL Dry Transfer Facility*				
	At ANL-W		*At SRS*						
Alternative	*EBR-II Driver*	*Fermi-1 Blanket*	*Declad EBR-II Blanket*	*Declad Fermi-1 Blanket*	*Ceramic Waste Form*	*Metallic Waste Form*	*Spent Fuel*	*Melt and Dilute Product*	*Total*
No Action	43	14	—	—	15	1	355	—	428
1	43	14	—	—	125	5	—	—	187
2	43	14	—	—	27	2	63	—	149
3	43	14	11	18	27	2	—	—	115
4	43	14	—	—	27	2	—	114	200
5	43	14	11	18	27	2	—	—	115
6	43	14	—	—	32	1	—	164	254

G.5.2 Representative Routes

Representative overland truck routes were selected for the shipments from ANL-W to SRS. The routes were selected consistent with current routing practices and all applicable routing regulations and guidelines (DOT 1992). However, the routes were determined for risk assessment purposes. They do not necessarily represent the actual routes that would be used to transport spent nuclear fuel in the future. Specific routes cannot be identified in advance. The representative truck routes are shown in **Figure G–5**.

Route characteristics that are important to the radiological risk assessment include the total shipment distance 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 **Table G–3**. The population densities along each route are derived from 1990 U.S. Bureau of Census data. Rural, suburban, and urban areas are characterized according to the following breakdown: rural population densities range from 0 to 54 persons per square kilometer (0 to 139 persons per square mile); the suburban range is from 55 to 1,284 persons per square kilometer (140 to 3,326 persons per square mile); and the urban range includes all population densities greater than 1,284 persons per square kilometer (3,326 persons per square mile). The affected population includes all persons living within 800 meters (0.5 mile) of each side of the road. The affected population, for route characterization and incident-free dose calculation, includes all persons living within 800 meters (0.5 mile) of each side of the road.

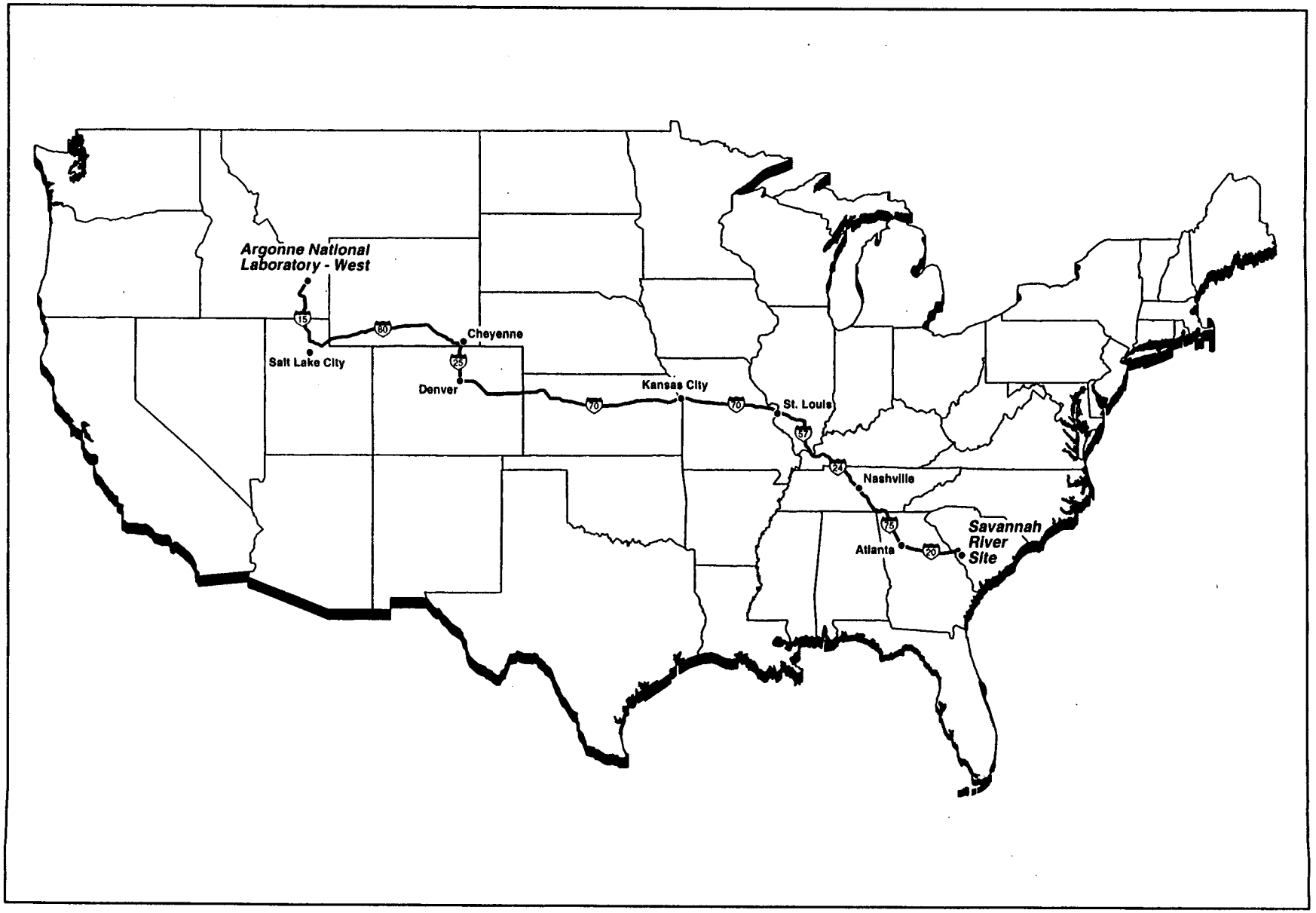


Figure G–5 Representative Overland Truck Route

Table G–3 Potential Shipping Routes Evaluated for the Sodium Bonded Spent Nuclear Fuel EIS

From	To	Distance (kilometers)	Percentages in Zones			Population Density in Zone (per square kilometer)			Number of Affected Persons
			Rural	Suburban	Urban	Rural	Suburban	Urban	
Truck Routes									
ANL-W	SRS	3,759.3	82.8	15.4	1.8	7.4	353	2,173.3	599,000
INTEC	ANL-W	38.6	100	0	0	1.0	N/A	N/A	62

N/A = not applicable.
The shipment impact to SRS are all based on the distance and population exposed on a trip from ANL-W to SRS.

G.5.3 External Dose Rates

External dose rates are calculated for the spent nuclear fuel being shipped on public roads (SAIC 1999). For the EBR-II blanket fuel, the dose rate on contact with the cask is 0.6 millirem per hour and the dose rate at 2 meters (6 feet) from the cask is 0.1 millirem per hour. For the Fermi-1 blanket fuel, the dose rate on contact with the cask is 0.00071 millirem per hour and the dose rate at 2 meters (6 feet) from the cask is 0.00014 millirem per hour. For the EBR-II driver fuel shipped to ANL-W, the dose rate on contact with the cask is 0.59 millirem per hour and the dose rate at 2 meters (6 feet) from the cask is 0.12 millirem per hour.

External dose rates for the waste production canisters could not be calculated because the Type B cask has not been identified. Ceramic waste form, metallic waste form and melt and dilute product canisters are conservatively assumed to have a dose rate at 2 meters (6 feet) from the vehicle equal to the maximum regulatory limit of 10 millirem per hour. The spent nuclear fuel waste is conservatively estimated to have a dose rate of 0.5 millirem per hour at 2 meters (6 feet) from the cask.

G.5.4 Health Risk Conversion Factors

The health risk conversion factors used to estimate expected cancer fatalities were: 0.0005 and 0.0004 latent cancer fatalities per person-rem for members of the public and workers, respectively (NCRP 1993).

G.5.5 Accident Frequencies

For the calculation of accident risks, vehicle accident and fatality rates are taken from data provided in other reports (ANL 1994). Accident rates are generically defined as the number of accident involvements (or fatalities) in a given year per unit of travel 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 in truck-kilometers) 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. 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 from 1986 to 1988 compiled by the U.S. Department of Transportation Office of Motor Carriers. Saricks and Kvitek (ANL 1994) present accident involvement and fatality counts; estimated kilometers of travel by state; and the corresponding average accident involvement, fatality, and injury rates for the three years investigated. A fatality caused by an accident is the death of a member of the public who is killed instantly or dies within 30 days due to the injuries sustained in the accident.

G.5.6 Container Accident Response Characteristics and Release Fractions

G.5.6.1 Development of Conditional Probabilities

NUREG-0170 (NRC 1977) originally was used to estimate the conditional probabilities associated with the accidents involving transportation of radioactive materials. The Modal Study, an initiative taken by the NRC (NRC 1987) to refine more precisely the analysis presented in NUREG-0170 for spent nuclear fuel shipping casks, was used to estimate the conditional probabilities of accidents.

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 assumed to have been designed, manufactured, operated, and maintained according to national codes and standards. Design parameters of the representative casks were chosen to meet the minimum test criteria specified in 10 CFR 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 region 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.

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. Although regions span the entire range of mechanical and thermal accident loads, they are grouped into accident categories that can be characterized by a single set of release fractions and are, therefore, considered together in the accident consequence assessment. The accident category severity fraction is the sum of all conditional probabilities in that accident category.

G.5.6.2 Release Fraction Assumptions

The release fractions were taken from the Programmatic Spent Nuclear Fuel EIS (DOE 1995), which was based on the above described Modal Study. Spent nuclear fuel could be shipped in two different forms: unaltered or declad. The construction and cladding of the spent nuclear fuel are assumed to be similar enough to the aluminum-clad fuel analyzed in that EIS that the performance in an accident would be similar. The declad fuel would also exhibit similar performance, since the fuel is placed in a shipping can which is in turn placed inside the transportation cask.

G.5.7 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 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 (Neuhauser and Kanipe 1998). These risks are 1×10^{-7} mortality per kilometer (1.6×10^{-7} per

mile) of truck travel in urban areas. 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 latent cancer fatalities. Vehicle-related risks from incident-free transportation (affecting the population in urban areas along the transportation route) 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 case. This method has been used in several EISs 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.

G.6 Risk Analysis Results

Per-shipment risk factors have been calculated for the collective populations of exposed persons and for the crew for all anticipated routes and shipment configurations. The radiological risks are presented in doses per shipment for each unique route, material, and container combination. The radiological dose per shipment factors for incident-free transportation are presented in **Table G–4** for the transportation routes analyzed for this EIS. For spent nuclear fuel to be transferred to INEEL, consistent with the Record of Decision for the Programmatic Spent Nuclear Fuel EIS, the following analysis is performed. As stated in Section G.5.1, the Programmatic Spent Nuclear Fuel EIS (DOE 1995) used very conservative assumptions to analyze the shipments from the Oak Ridge Reservation, Hanford Site, and Sandia National Laboratory/New Mexico. For these 12 shipments, the incident free public risk is 0.00097 latent cancer fatalities from radiation and 8.1×10^{-6} latent cancer fatalities from exhaust emissions. The crew radiological risk is 0.00031 cancer fatalities. The public risk from radiological accidents is 0.00004 latent cancer fatalities and from nonradiological accidents is 0.0012 fatalities.

Doses are calculated for the crew, off-link public (i.e., people living along the route), on-link public (i.e., pedestrians and drivers along the route), and public at rest and fueling stops (i.e., stopped cars, buses and trucks, workers, and other bystanders). For the onsite shipments from INTEC to ANL-W, the stop dose is set to zero, because a truck would not be expected to stop during a trip that takes less than an hour. The off-link dose is zero because no persons are residing within 800 meters (0.5 miles) of the road.

The radiological dose risk factors for transportation accidents are also presented in Table G–2. The accident risk factors are called "dose risk" because the values incorporate the spectrum of accident severity probabilities and associated consequences. The accident dose is very low because, although persons are residing in an 80 kilometers (50 miles) radius of the road, they are generally quite far from the road. Since RADTRAN 5 uses an assumption of homogeneous population from the road out to 80 kilometers (50 miles), it would greatly overestimate the actual doses. However, the doses are clearly several factors of ten lower than the doses for the other transportation legs shown in Table G–4.

The nonradiological risk factors are presented in fatalities per shipment in **Table G–5**. Separate risk factors are provided for fatalities resulting from exhaust emissions (caused by hydrocarbon emissions known to be carcinogens) and transportation accidents (fatalities resulting from impact).

Table G–6 shows the risks of transportation for each alternative. The risks are calculated by multiplying the previously given per-shipment factors by the number of shipments over the duration of the program and, for the radiological doses, by the health risk conversion factors.

Table G–4 Radiological Risk Factors for Single Shipments

From	To	Material and Package	Incident-Free Dose (person-rem)					Accident Dose (person-rem)
			Crew	Public				
				Off-Link	On-Link	Stops	Total	
ANL	SRS	EBR-II blanket	0.000107	0.000174	0.000902	3.25×10^{-7}	0.00108	2.71×10^{-7}
ANL	SRS	Fermi-1 blanket	1.34×10^{-7}	2.18×10^{-7}	1.13×10^{-6}	4.06×10^{-10}	1.35×10^{-6}	3.55×10^{-9}
INTEC	ANL-W	EBR-II driver	1.10×10^{-6}	0	8.10×10^{-6}	0	8.10×10^{-6}	less than 1×10^{-10}
ANL-W	INEEL	Ceramic waste - driver	0.000137	0	0.00101	0	0.00101	less than 1×10^{-10}
ANL-W	INEEL	Ceramic waste - blanket	4.12×10^{-6}	0	0.0000304	0	0.0000304	less than 1×10^{-10}
ANL-W	INEEL	Metallic waste - driver	0.000137	0	0.00101	0	0.00101	less than 1×10^{-10}
ANL-W	INEEL	Metallic waste - blanket	4.12×10^{-6}	0	0.0000304	0	0.0000304	less than 1×10^{-10}
ANL-W	INEEL	Melt and dilute waste - driver	0.000137	0	0.00101	0	0.00101	less than 1×10^{-10}
ANL-W	INEEL	Melt and dilute waste - blanket	0.000137	0	0.00101	0	0.00101	less than 1×10^{-10}
ANL-W	INEEL	Spent fuel	4.12×10^{-6}	0	0.0000304	0	0.0000304	less than 1×10^{-10}

Table G–5 Nonradiological Risk Factors per Shipment

Nonradiological Risk Estimates (fatalities/shipment)			
From	To	Exhaust Emission	Accident
ANL-W	SRS	6.8×10^{-6}	0.000030
INTEC	ANL-W	0	3.0×10^{-7}

Table G–6 Risks of Transporting the Hazardous Materials[a]

Material Shipped[b]	Alternative	Distance on Public Roads (kilometers)	Incident-Free			Accident	
			Radiological		Nonradiological		
			Crew	Public	Emission	Traffic	Radiological
EBR-II driver and Fermi-1 blanket fuel	No Action	15,980	1.22×10^{-6}	0.000011	0	0.00025	less than 1×10^{-9}
EBR-II driver and Fermi-1 blanket fuel	1	6,678	1.77×10^{-6}	0.000016	0	0.00010	less than 1×10^{-9}
EBR-II driver and Fermi-1 blanket fuel	2	5,211	1.71×10^{-6}	0.000016	0	0.00008	less than 1×10^{-9}
EBR-II driver and declad and cleaned EBR-II and Fermi-1 blanket fuel	3	111,799	2.08×10^{-6}	0.000021	0.00039	0.0018	1.7×10^{-9}
	3 (SRS)	109,020	4.7×10^{-7}	0.000006	0.00039	0.0017	1.5×10^{-9}
	3 (ANL-W)	2,779	1.6×10^{-6}	0.000015	0	0.000045	less than 10^{-9}
EBR-II driver and Fermi-1 blanket fuel	4	7,180	7.86×10^{-6}	0.000072	0	0.00011	less than 1×10^{-9}
EBR-II driver and declad and cleaned EBR-II and Fermi-1 blanket fuel	5[c]	111,799	2.08×10^{-6}	0.000021	0.00039	0.0018	1.7×10^{-9}
EBR-II driver and Fermi-1 blanket fuel	6	9,264	0.000011	0.00010	0	0.00014	less than 1×10^{-9}

[a] All risks are expressed as number of latent cancer fatalities, except for the Accident-Traffic column, which lists number of accident fatalities.
[b] Also includes shipments of ceramic and metallic high-level radioactive waste under all alternatives.
[c] For details on breakdown of risk, see the values given for Alternative 3.

The risks to various exposed individuals under incident-free transportation conditions have been estimated for hypothetical exposure scenarios. The estimated doses to workers and the public are presented in **Table G–7**.

Table G–7 Estimated Dose to Exposed Individuals During Incident-Free Transportation Conditions

Receptor		*Dose to Maximally Exposed Individual*	
		Idaho to SRS	***Intrasite***
Workers	Crew member (truck driver) *	0.00008 rem per year	0.002 rem per year
	Inspector	0.000029 rem per event	Not applicable
Public	Resident	4.0×10^{-9} rem per event	Not applicable
	Person in traffic congestion	0.00011 rem per event	0.003 rem per event
	Person at service station	0.00001 rem per event	Not applicable

* Assumes that an individual driver takes every shipment.

All doses are presented on a per-event basis (person-rem per event) because it is not likely that the same person will be exposed to multiple events. The maximum dose to a crew member is based on the same individual being responsible for driving every shipment for the duration of the campaign. Note that the potential exists for larger individual exposures if multiple exposure events occur. For example, the dose to a person stuck in traffic next to a shipment for 10 minutes is calculated to be 0.03 millirem. However, since the intersite shipments pass through urban areas, a 30-minute exposure time is considered. Using the estimated dose rates, the maximally exposed individual would receive 0.1 millirem. The onsite shipments have a higher dose rate, but the maximum time stuck in traffic next to the waste shipment is considered to be 10 minutes. If the exposure duration were longer, the dose would rise proportionally. In addition, a person working at a truck service station could receive a significant dose if trucks were to use the same stops repeatedly. The dose to a person fueling a truck could be as much as 0.01 millirem per event.

The cumulative dose to a resident was calculated assuming all shipments passed his or her home. The cumulative doses assume that the resident is present for every shipment and is unshielded at a distance of 30 meters (about 98 feet) from the route. Therefore, the cumulative dose depends on the number of shipments passing a particular point and is independent of the actual route being considered. The maximum dose to this resident, if all the material were to be shipped via this route, would be less than 0.01 millirem.

The estimated dose to transportation crew members is presented for a commercial crew. No credit is taken for the shielding associated with the tractor or trailer.

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 shipment. The maximum foreseeable (frequency greater than 1×10^{-7} per year) offsite transportation accident involves a shipment of EBR-II blanket fuel material under neutral (average) weather conditions. The accident has a probability of occurrence of about 1 every 10 million years and could result in 0.46 person-rem to the public. Additionally the accident could result in a dose of 0.0019 rem to the hypothetical maximally exposed individual in the immediate vicinity of the accident. The probability of an accident occurring and the exposed populations are lower for the onsite shipment of EBR-II blanket fuel. The source term is lower for the offsite shipments of Fermi blanket fuel. This accident would fall into Severity Category 5 of the Modal Study accident matrix (NRC 1987), and would occur in a suburban population zone. To incur this level of damage, the cask would have to collide with an immovable object at a speed of much greater than 88 kilometers per hour (55 miles per hour). The probability of an accident with a more energetic collision or a significant fire, which could lead to higher consequences, is lower.

G.7 CONCLUSIONS AND LONG-TERM IMPACTS OF TRANSPORTATION

G.7.1 Conclusions

It is unlikely that the transportation of radioactive materials will cause an additional fatality.

G.7.2 Long-Term Impacts of Transportation

The Programmatic Spent Nuclear Fuel EIS (DOE 1995) analyzed the cumulative impacts of all transportation of radioactive materials, including impacts from reasonably foreseeable actions that include transportation of radioactive material for a specific purpose and general radioactive materials transportation that is not related to a particular action. The total worker and general population collective doses are summarized in **Table G–8**. The table shows that the impacts of this program are quite small compared with overall transportation impacts. Total collective worker dose from all types of shipments (historical, the alternatives, reasonably foreseeable actions, and general transportation) was estimated to be 320,000 person-rem (130 latent cancer fatalities) for the period 1943 through 2035 (93 years). Total general population collective dose was also estimated to be 320,000 person-rem (160 latent cancer fatalities). 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 shipments of radiopharmaceuticals to nuclear medicine laboratories and shipments of commercial low-level radioactive waste to commercial disposal facilities. The total number of latent cancer fatalities estimated to result from radioactive materials transportation over the period between 1943 and 2035 was 290. Over this same period (93 years), approximately 28 million people would die from cancer, based on 300,000 cancer fatalities per year. It should be noted that the estimated number of transportation-related latent cancer fatalities would be indistinguishable from other latent cancer fatalities, and the transportation-related latent cancer fatalities are 0.0010 percent of the total number of latent cancer fatalities.

Table G–8 Cumulative Transportation-Related Radiological Collective Doses and Latent Cancer Fatalities (1943 to 2035)

Category	*Collective Worker Dose (person-rem)*	*Collective General Population Dose (person-rem)*
Sodium-bonded spent nuclear fuel impacts (from Table G–4)	less than 1	less than 1
Other Nuclear Material Shipments		
Truck	11,000	50,000
Rail	820	1,700
General transportation (1943–2035)	310,000	270,000
Total collective dose	322,000	322,000
Total latent cancer fatalities	130	160

Source: DOE 1995.

G.8 UNCERTAINTY AND CONSERVATISM IN ESTIMATED IMPACTS

The sequence of analyses performed to generate the estimates of radiological risk for transportation 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 estimating 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 (e.g., 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 reality and conservatism of the assumption are addressed. Where practical, the parameters that most significantly affect the risk assessment results are identified.

G.8.1 Uncertainties in Material Inventory and Characterization

The inventories 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 dimensions of package contents, the strength of the radiation field, the heat that must be dissipated, and assumptions concerning shipment capacities. The physical and radiological characteristics are important in determining the material released during accidents and the subsequent doses to exposed individuals through multiple environmental exposure pathways.

Uncertainties in the inventory and characterization are reflected in the transportation risk results. If the inventory is overestimated (or underestimated), the resulting transportation risk estimates are also overestimated (or underestimated) by roughly the same factor. However, the same 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 the alternatives, as given in Table G-5, are believed to represent unbiased, reasonably accurate estimates from current information in terms of relative risk comparisons.

G.8.2 Uncertainties in Containers, Shipment Capacities, and Number of Shipments

The transportation required for each alternative is based in part on assumptions concerning the packaging characteristics and shipment capacities for commercial trucks. 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 such 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.

G.8.3 Uncertainties in Route Determination

Representative routes have been determined between all origin and destination sites considered in the EIS. The routes have been determined to be 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 concerning distances and total population along the routes. Moreover, since materials could be transported over an extended time starting at some time in the future, the highway 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. Specific routes cannot be identified in advance because the routes are classified to protect national security interests.

G.8.4 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. Estimating the accuracy or absolute uncertainty of the risk assessment results is generally difficult. 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 reduced by using state-of-the-art computer codes that have undergone extensive review. Because many uncertainties are recognized but difficult to quantify, assumptions are made at each step of the risk assessment process intended to produce conservative results (i.e., overestimate the calculated dose and radiological risk). Because parameters and assumptions are applied 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.

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 according to 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 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 doses, however, assume 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 about 50 percent of the total population dose for rural accidents. Interdiction of foodstuffs would act to reduce, but not eliminate, this contribution.

G.9 REFERENCES

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Appendix H

Environmental Justice Analysis

APPENDIX H
ENVIRONMENTAL JUSTICE ANALYSIS

H.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, the disproportionately high and adverse health or environmental effects of their programs, policies, and activities on minority populations and low-income populations.

The Council on Environmental Quality has oversight responsibility for documentation prepared in compliance with the National Environmental Policy Act (NEPA). In December 1997, the Council released its guidance on environmental justice under NEPA (CEQ 1997). The Council's guidance was adopted as the basis for the analysis of environmental justice contained in this environmental impact statement (EIS).

This appendix provides an assessment of the potential for disproportionately high and adverse human health or environmental effects on minority or low-income populations that could result from implementation of alternatives for management of the U.S. Department of Energy's (DOE) inventory of sodium-bonded spent nuclear fuel.

H.2 DEFINITIONS AND APPROACH

Minority Individuals and Population

The following definitions of minority individuals and population were used in this analysis of environmental justice:

- **Minority Individuals**—Members of any of the following population groups: Hispanic, Native American, Asian or Pacific Islander, or Black
- **Minority Population**—The total number of minority individuals residing within a potentially affected area

In discussions of environmental justice in this EIS, persons self-designated as Hispanic are included in the Hispanic population, regardless of race. For example, the Asian or Pacific Islander population is composed of persons self-designated as Asian or Pacific Islander and not of Hispanic origin. Asian or Pacific Islanders who designate themselves as having Hispanic origins are included in the Hispanic population. Data for the analysis of minorities and racial population were extracted for the year 2010 from the U.S. Census Bureau's worldwide web site (DOC 1999).

Executive Order 12898 specifically addresses "disproportionately high and adverse effects" on "low-income" populations. The Council on Environmental Quality recommends that poverty thresholds be used to identify "low-income" individuals (CEQ 1997).

Low-Income Individuals and Population

The following definitions of low-income individuals and population were used in this analysis:

- **Low-Income Individuals**—Persons whose self-reported incomes are less than the poverty threshold
- **Low-Income Population**—The total number of poverty-level individuals residing within a potentially affected area

Data for the analysis of low-income populations were extracted from the U.S. Census Bureau's Table P121 of Standard Tape File 3 (DOC 1992).

Disproportionately High and Adverse Human Health Effects

Adverse health effects are measured in risks and rates that could result in latent cancer fatalities, as well as other fatal or nonfatal adverse impacts to human health. Disproportionately high and adverse human health effects occur when the risk or rate of exposure to an environmental hazard for a minority or low-income population is significant and exceeds the risk of exposure rate for the general population or, where available, for another appropriate comparison group (CEQ 1997).

Disproportionately High and Adverse Environmental Impacts

A disproportionately high environmental impact refers to an impact or risk of an impact in a low-income or minority community that is significant and exceeds the environmental impact on the larger community. An adverse environmental impact is a deleterious environmental impact that is determined to be significant. In assessing cultural and aesthetic environmental impacts, impacts that uniquely affect geographically dislocated or dispersed low-income or minority populations were considered (CEQ 1997).

Potentially affected areas examined in this EIS include areas defined by an 80-kilometer (50-mile) radius centered on candidate facilities for the treatment and management of sodium-bonded spent nuclear fuel at Argonne National Laboratory-West (ANL-W) and the Savannah River Site (SRS).

H.3 Methodology

H.3.1 Spatial Resolution

For the purposes of enumeration and analysis, the U.S. Census Bureau has defined a variety of areal units (DOC 1992). Areal units of concern in this EIS include (in order of increasing spatial resolution) states, counties, census tracts, block groups, and blocks. The block is the smallest of these entities and offers the finest spatial resolution. This term refers to a relatively small geographical area bounded on all sides by visible features such as streets and streams or by invisible boundaries such as city limits and property lines. During the 1990 census, the U.S. Census Bureau subdivided the United States and its territories into 7,017,425 blocks. For comparison, the number of counties, census tracts, and block groups used in the 1990 census were 3,248; 62,276; and 229,192, respectively. While blocks offer the finest spatial resolution, economic data required for identification of low-income populations are not available at the block level of spatial resolution. In the analysis below, block groups are used throughout as the areal unit. Block groups generally contain between 250 and 500 housing units (DOC 1992).

During the decennial census, the U.S. Census Bureau collects data from individuals and aggregates the data according to residence in a geographical area, such as a county or block group. Boundaries of the areal units are selected to coincide with features such as streams and roads or political boundaries such as county and city borders. Boundaries used for aggregation of the census data usually do not coincide with boundaries used in the calculation of health effects. As discussed in Chapter 4 of this EIS, radiological health effects due to an accident at each of the sites are evaluated for persons residing within a distance of 80 kilometers (50 miles) of the accident site. In general, the boundary of the circle with an 80-kilometer (50-mile) radius centered at the accident site will not coincide with boundaries used by the U.S. Census Bureau for enumeration of the population in the potentially affected area. Some block groups lie completely inside or outside of the radius for health effects calculation. However, other block groups are only partially included. As a result of these partial inclusions, uncertainties are introduced into the estimate of the population at risk from the accident.

To estimate the populations at risk in partially included block groups, it was assumed that populations are uniformly distributed throughout the area of each block group. For example, if 30 percent of the area of a block group lies within 80 kilometers (50 miles) of the accident site, it was assumed that 30 percent of the population residing in that block group would be at risk. An upper bound for the population at risk was obtained by including the total population of partially included block groups in the population at risk. Similarly, a lower bound for the population at risk was obtained by excluding the population of partially included blocks from the population at risk. As a general rule, if the areas of geographic units defined by the U.S. Census Bureau are small in comparison with the potentially affected area, then the uncertainties due to partial inclusions will be relatively small.

Tables H-1 through **H-3** show lower and upper population bounds for the ANL-W site located within the boundary of the Idaho National Engineering and Environmental Laboratory (INEEL) in Idaho, and F- and L-Areas at SRS in South Carolina. Estimated populations listed in column 3 of these tables were obtained under the assumption that populations are distributed uniformly throughout each block group that lies at least partly within 80 kilometers (50 miles) of the candidate sites. Lower population bounds given in column 2 were obtained by summing only those populations residing in block groups that are wholly included within a circle of an 80-kilometer radius centered at each candidate site. Upper bounds shown in column 4 are the sum of populations residing within all block groups that are at least partly included within that circle. For these candidate sites, lower bounds differed from the corresponding estimate by no more than 12.9 percent, while upper bounds differed from the corresponding estimate by 10.4 percent or less. As discussed in Chapter 4 and summarized in Table 2-4 of Chapter 2, implementation of the alternatives would pose no significant radiological or nonradiological risks to the general public. Under normal operations, the radiological risk of a latent cancer fatality among the surrounding population is approximately one in 90,000 or less. In the event of an accident involving a radiological release affecting the general population, the maximum risk to the public would occur at SRS under Alternative 5, where 0.013 latent cancer fatalities would be expected. Under Alternative 5, unless the population at risk near SRS were increased by nearly a factor of 77 over the estimated value, no latent cancer fatalities would be expected. Thus, uncertainties in the estimates of total, minority, and low-income populations are not large enough to noticeably affect the conclusions regarding environmental effects on minority and low-income populations that would result from implementation of the proposed action or alternatives.

Table H–1 Total Population Estimates and Bounds in 1990 for Candidate Sites

Candidate Site	*Lower Bound*	*Estimated Total Population*	*Upper Bound*
ANL-W	168,365	181,088	197,519
F-Area	569,693	608,891	660,363
L-Area	559,870	606,819	663,376

Table H–2 Minority Population Estimates and Bounds in 1990 for Candidate Sites

Candidate Site	*Lower Bound*	*Estimated Minority Population*	*Upper Bound*
ANL-W	13,712	15,737	17,369
F-Area	215,781	230,116	251,696
L-Area	218,414	237,094	260,629

Table H–3 Low-Income Population Estimates and Bounds in 1990 for Candidate Sites

Candidate Site	*Lower Bound*	*Estimated Low-Income Population*	*Upper Bound*
ANL-W	22,966	25,105	27,455
F-Area	98,972	106,281	116,037
L-Area	98,519	107,469	117,988

H.3.2 Population Projections

Health effects were calculated for populations projected to reside in potentially affected areas during the year 2010. Extrapolations of the total population for individual states are available from both the U.S. Census Bureau and various state agencies (Campbell 1996). The U.S. Census Bureau also projects populations by ethnic and racial classification in one-year intervals for the years from 1995 to 2025 at the state level. State agencies project total populations for individual counties. No Federal or state agency projects block groups or low-income populations. Data used to project minority populations were extracted from the U.S. Census Bureau's Internet web site (DOC 1999). To project minority populations in potentially affected areas, minority populations determined from the 1990 census data were taken as a baseline for each block group. Then it was assumed that percentage changes in the minority population of each block group for a given year (compared to the 1990 baseline data) will be the same as percentage changes in the state minority population projected for the same year. An advantage to this assumption is that the projected populations are obtained using a consistent method, regardless of the state and associated block group involved in the calculation. A disadvantage is that the method is insensitive to localized demographic changes that could alter the projection in a specific area.

The U.S. Census Bureau uses the cohort-component method to estimate future populations for each state (Campbell 1996). The set of cohorts is composed of: (1) age groups from 1 year or less to 85 years or more, (2) male and female populations in each age group, and (3) the following racial and ethnic groups in each age group: Hispanic, non-Hispanic Asian, non-Hispanic African American, non-Hispanic Native American, and non-Hispanic White. Components of the population change used in the demographic accounting system are births, deaths, net state-to-state migration, and net international migration. If P(t) denotes the number of individuals in a given cohort at time "t," then:

$$P(t) = P(t_0) + B - D + DIM - DOM + IIM - IOM \quad (1)$$

where:

$P(t_0)$	=	Cohort population at time t_0 is less than or equal to t. For this analysis, t_0 denotes the year 1990.
B	=	Births expected during the period from t_0 to t.
D	=	Deaths expected during the period from t_0 to t.
DIM	=	Domestic migration into the state expected during the period from t_0 to t.
DOM	=	Domestic migration out of the state expected during the period from t_0 to t.
IIM	=	International migration into the state expected during the period from t_0 to t.
IOM	=	International migration out of the state expected during the period from t_0 to t.

Estimated values for the components shown on the right side of the equation are based on past data and various assumptions regarding changes in the rates for birth, mortality, and migration (Campbell 1996). Persons of Hispanic origin are included in the Hispanic population regardless of race. It should be noted that the U.S. Census Bureau does not project populations of individuals who identified themselves as "other race" during the 1990 census. This population group is less than 2 percent of the total population in each of the states. However, to project total populations in the environmental justice analysis, population projections for the "other race" group were made under the assumption that the growth rate for the "other race" population will be identical to the growth rate for the combined minority and White populations.

H.4 Environmental Justice Assessment

The analysis of environmental justice effects was based on an assessment of the impacts reported in Chapter 4 of this EIS. This analysis was performed to identify any disproportionately high and adverse human health or environmental impacts on minority or low-income populations surrounding ANL-W and SRS. Demographic information obtained from the U.S. Census Bureau was used to identify the minority populations and low-income communities in the zone of potential impact surrounding the two sites. The zone, or region of influence, is a circle that has an 80-kilometer (50-mile) radius around the proposed sites. This radius is consistent with that used to evaluate the collective dose for human health effects, air impact modeling, and socioeconomic impacts, and is judged to encompass all of the impacts that may occur.

H.5 Results for the Sites

As discussed in Chapter 2 of this EIS, candidate sites for the treatment and management of sodium-bonded spent nuclear fuel are located at ANL-W and SRS. This section describes the environmental justice analysis of potentially affected minority and low-income populations residing near the candidate sites. It should be noted that projections of the total population provided in this appendix differ from the projected total populations used in the health effects calculations described in Chapter 4. This is because the projections used in the analysis of environmental justice are based on projections for the states provided by the U.S. Bureau of the Census (Campbell 1996). Projections used in the analysis of health effects are based on county-wide projections provided by state agencies. As discussed in Section H.3.2, the county projections are more sensitive to localized demographic changes. However, the states do not provide projections for minority populations. Therefore, the U.S. Bureau of the Census projections were used in the analysis of environmental justice. Population projections obtained with the two approaches differ by 8 percent or less and have essentially no effect on these results of the analyses.

H.5.1 Argonne National Laboratory-West

Figure H–1 shows the racial and ethnic composition of the minority population of ANL-W projected to reside in the potentially affected area in the year 2010. In the interval between 1990 and 2010, the percentage of the total population composed of minorities is projected to increase from 8.7 percent to 13.3 percent. For comparison, during the 1990 census, minorities were found to compose approximately one-quarter of the total national population. By the year 2010, minorities are projected to compose closer to one-third of the total national population. The percentage of the minority population residing in the potentially affected area surrounding ANL-W was less than the corresponding national percentage in 1990, and is expected to remain so through the year 2010. Hispanics are the largest minority group residing in the potentially affected area, while the Asian and Hispanic populations are projected to show the largest growth rates.

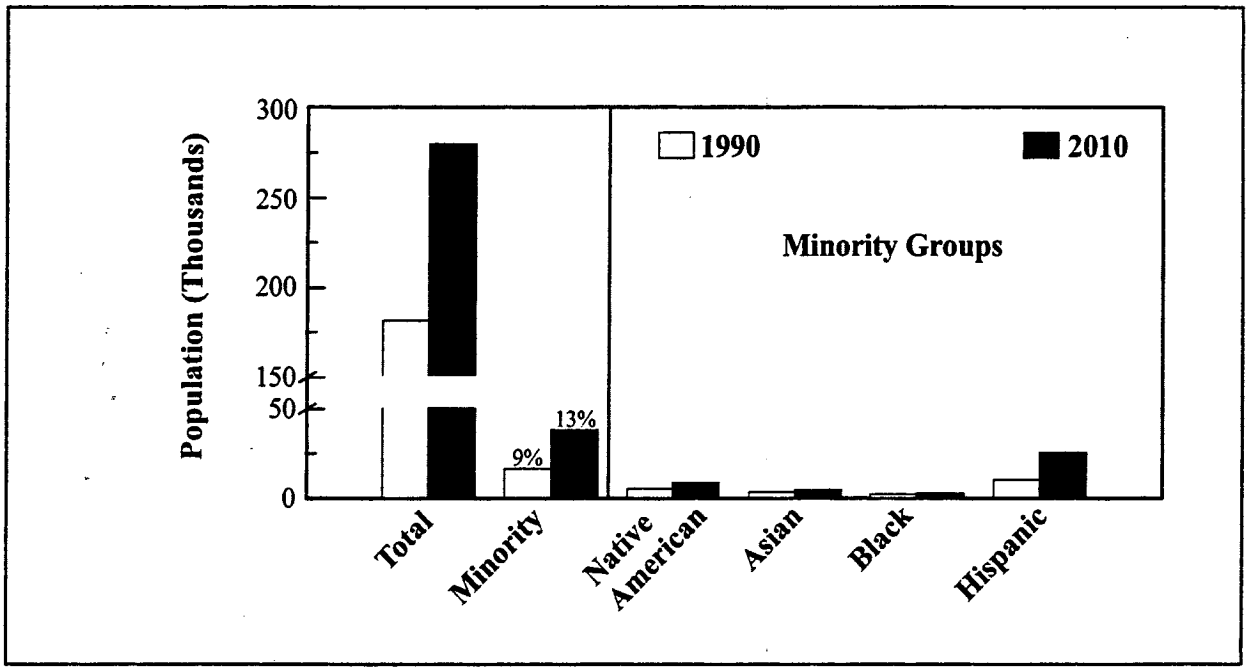


Figure H–1 Projected Racial and Ethnic Composition of the Minority Population Residing Within 80 Kilometers (50 Miles) of ANL-W in 2010

Figure H–2 shows the location of minority populations residing near the ANL-W in 1990. As indicated in the figure, block groups for which the percentage of minority residents exceeds the corresponding national percentage are located throughout the potentially affected area.

During the 1990 census, 15 percent of the residents within the potentially affected area surrounding ANL-W reported incomes below the poverty threshold. Slightly over 13 percent of the national population reported incomes below the poverty threshold, and approximately 13 percent of the residents of Idaho reported incomes below the poverty threshold during the same year. Thus, the percentage of the low-income population residing within the potentially affected area exceeded that for the nation and the state of Idaho by approximately 2 percent. **Figure H–3** shows the geographical distribution of low-income residents surrounding the ANL-W site in 1990. Block groups for which the percentage of low-income residents exceeds the corresponding national percentage are located throughout the potentially affected area.

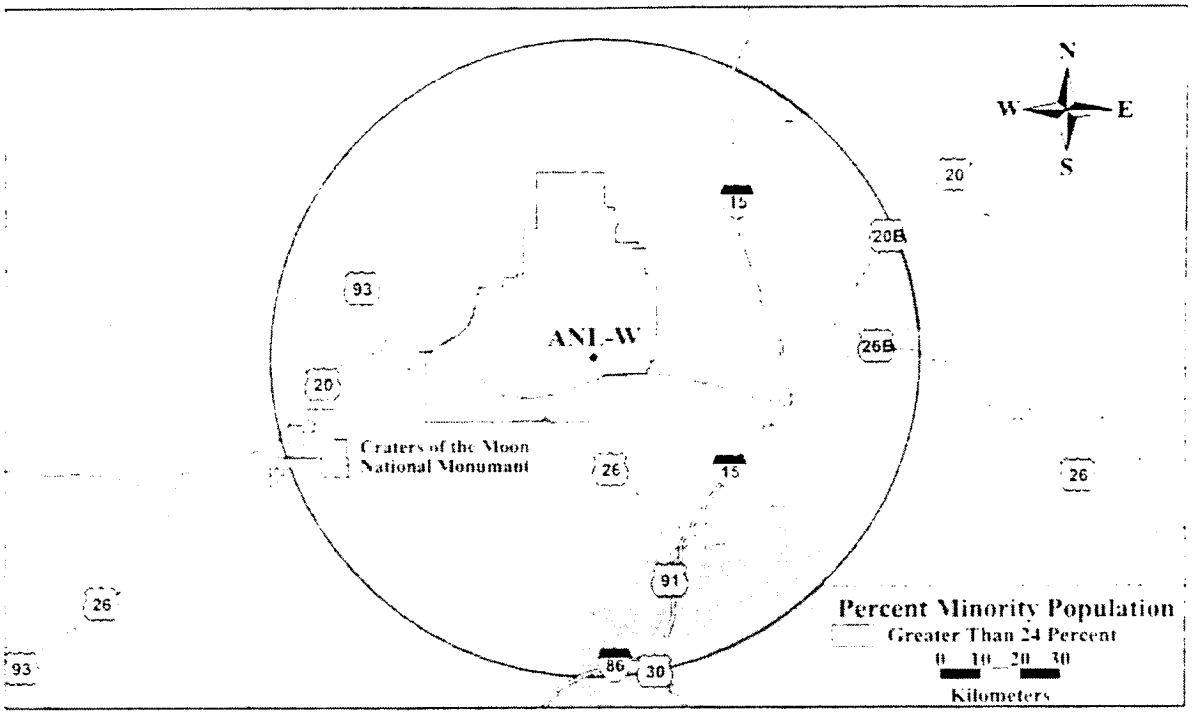


Figure H-2 Minority Population Residing Within 80 Kilometers (50 Miles) of the ANL-W Site in 1990

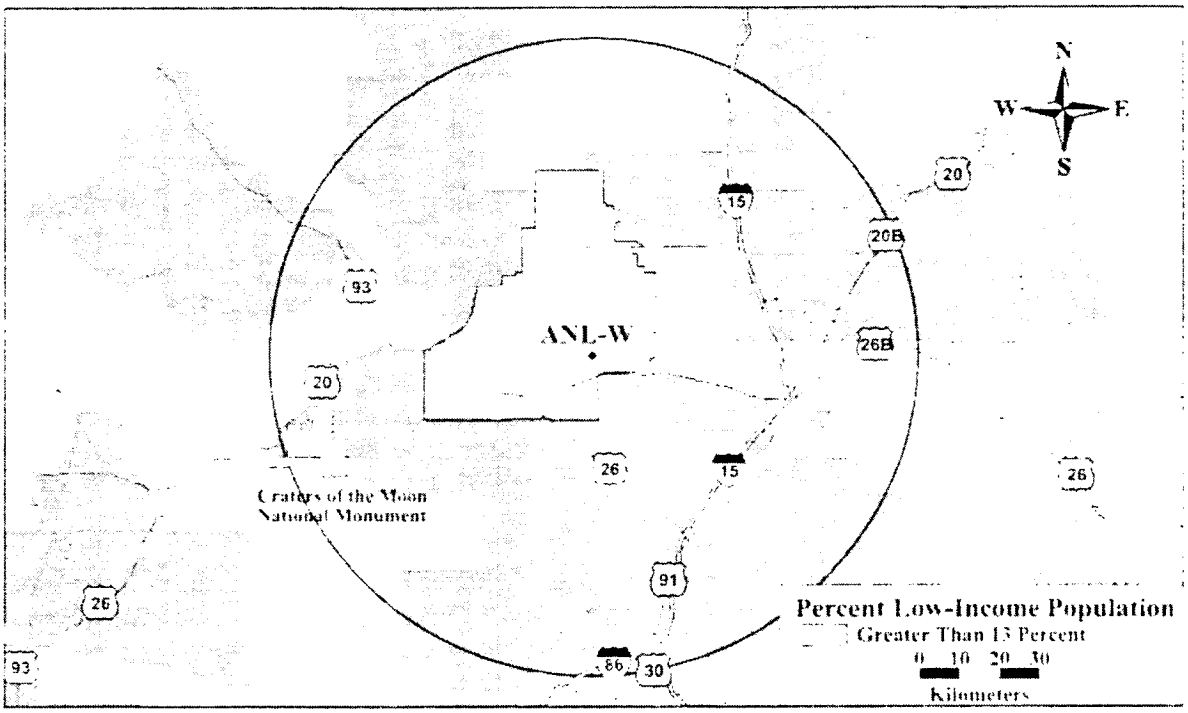


Figure H-3 Low-Income Population Residing Within 80 Kilometers (50 Miles) of ANL-W in 1990

H.5.2 The Savannah River Site F-Area

Figure H–4 shows the racial and ethnic composition of the minority population residing within 80 kilometers (50 miles) of F-Area at SRS in 1990, and the minority population projected to reside in the potentially affected area in the year 2010. In the interval between 1990 and 2010, the percentage of the total population composed of minorities is projected to increase from 37.9 percent to 42 percent. For comparison, during the 1990 census, minorities were found to compose approximately one-quarter of the total national population. By the year 2010, minorities are projected to compose nearly one-third of the total national population. The percentage of the minority population residing in the potentially affected area surrounding F-Area was larger than the corresponding national percentage in 1990, and is expected to remain so through the year 2010. Blacks are the largest minority group residing in the potentially affected area, while the Asian and Hispanic populations are projected to show the largest growth rates.

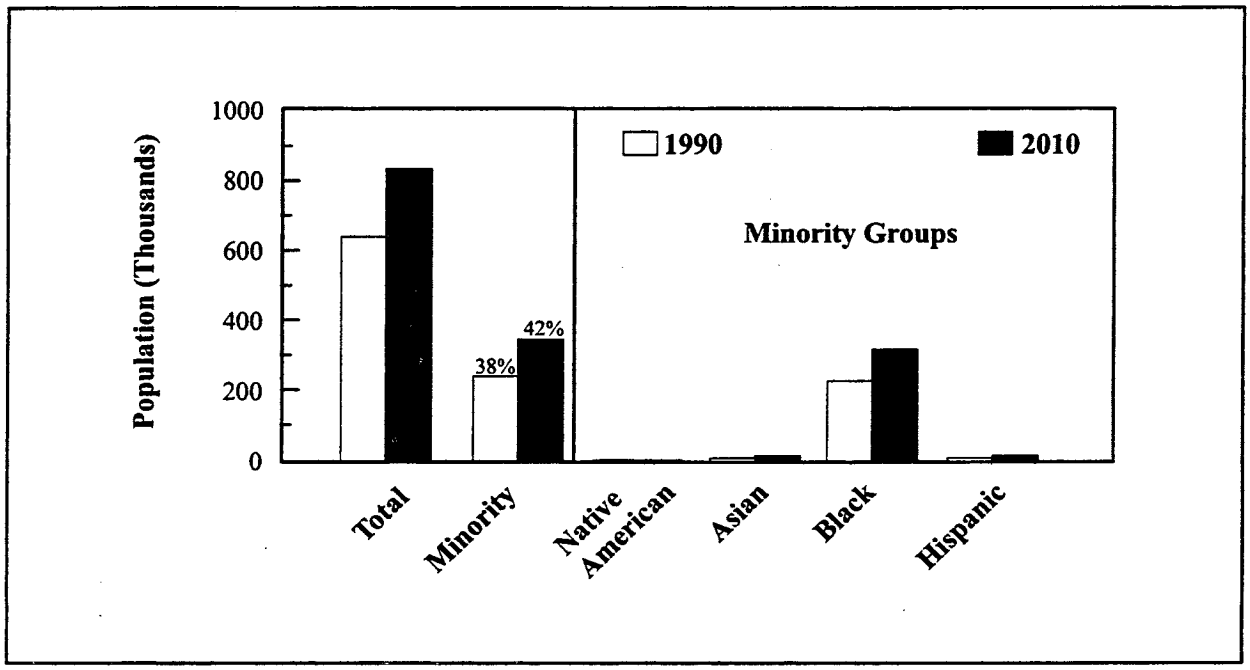


Figure H-4 Racial and Ethnic Composition of the Minority Population Residing Within 80 Kilometers (50 Miles) of the SRS F-Area in 2010

Figure H–5 shows the geographical distribution of minority populations residing near the SRS F-Area (and L-Area) in 1990. L-Area is discussed in Section H.5.3, below. Block groups for which the percentage of the minority population exceeds the national percentage are located throughout the potentially affected area surrounding F-Area.

During the 1990 census, 18 percent of the residents within the potentially affected area surrounding F-Area reported incomes below the poverty threshold. Slightly over 13 percent of the national population reported incomes below the poverty threshold, and nearly 15 percent of the residents of the combined States of Georgia and South Carolina reported incomes below the poverty threshold during the same year. Thus, the percentage of low-income population residing within the potentially affected area exceeded that for the Nation and the States of Georgia and South Carolina. **Figure H–6** shows the geographical distribution of low-income residents surrounding the F-Area site (and L-Area Site) in 1990. Block groups for which the percentage of low-income residents exceeds the corresponding national percentage are located throughout the potentially affected area.

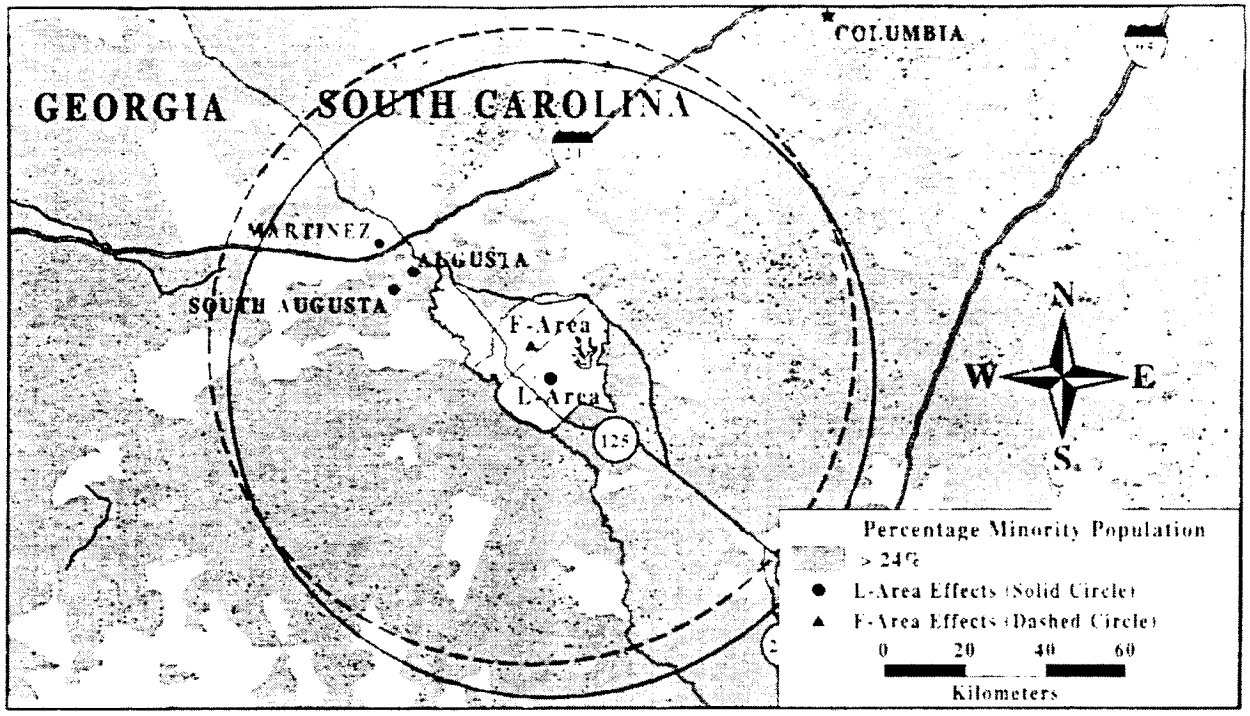


Figure H–5 Minority Population Residing Within 80 Kilometers (50 Miles) of SRS F-Area and L-Area in 1990

H.5.3 The Savannah River Site L-Area

Figure H–7 shows the racial and ethnic composition of the minority population projected to reside in the potentially affected area surrounding the SRS L-Area by the year 2010. In the interval between 1990 and 2010, the percentage of the total population composed of minorities is projected to increase from 39.1 percent to 43 percent. For comparison, during the 1990 census, minorities were found to compose approximately one-quarter of the total national population. By the year 2010, minorities are projected to compose close to one-third of the total national population. The percentage of the minority population residing in the potentially affected area surrounding L-Area was larger than the corresponding national percentage in 1990, and is expected to remain so through the year 2010. Blacks are the largest minority group residing in the potentially affected area, while the Asian and Hispanic populations are projected to show the largest growth rates.

Figure H–5 shows the geographical distribution of minority populations residing near the SRS L-Area and F-Area in 1990. F-Area was discussed in Section H.5.2 above. As indicated in the figure, block groups for which the percentage of minority residents exceeds the national percentage are distributed throughout the potentially affected area surrounding L-Area.

During the 1990 census, 20.6 percent of the residents within the potentially affected area surrounding L-Area reported incomes below the poverty threshold. Slightly over 13 percent of the national population reported incomes below the poverty threshold, and nearly 15 percent of the residents of the combined States of Georgia and South Carolina reported incomes below the poverty threshold during the same year. Thus, the percentage low-income population residing within the potentially affected area exceeded that for the Nation and the States of Georgia and South Carolina. As shown in Figure H–6, block groups for which the percentage of low-income residents exceeds the corresponding national percentage are located throughout the potentially affected area.

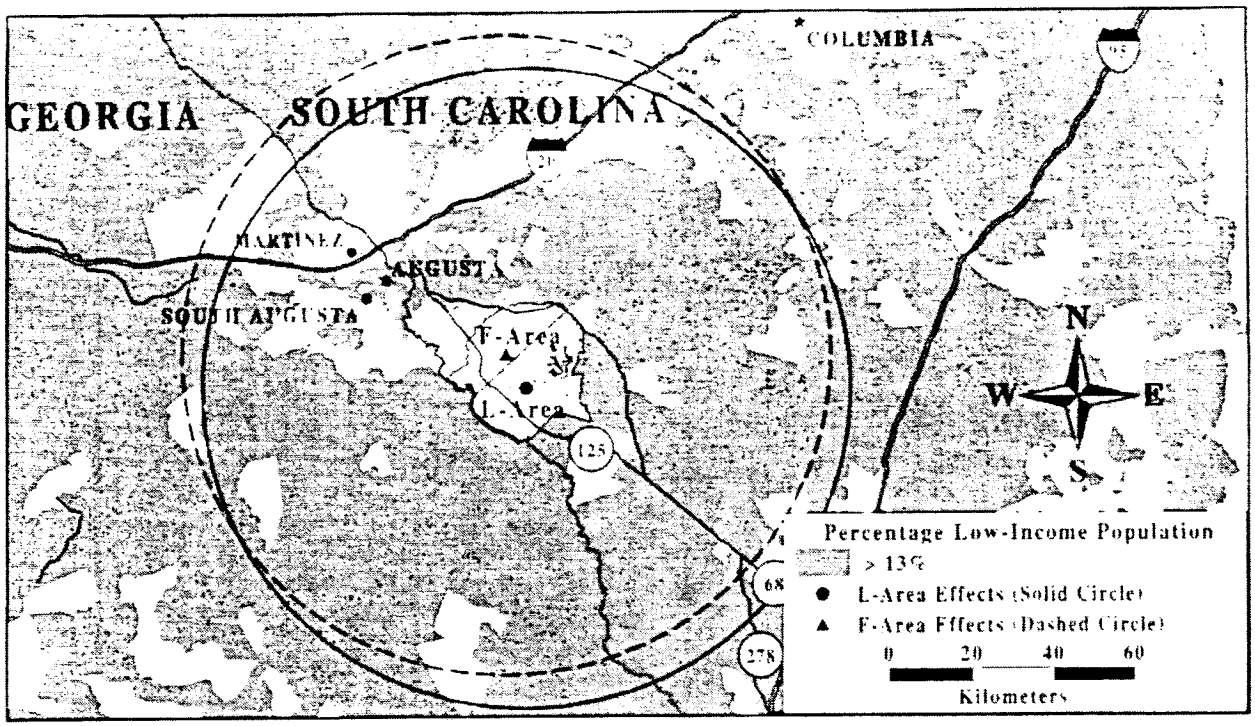


Figure H-6 Low-Income Populations Residing Within 80 Kilometers (50 Miles) of SRS F-Area and L-Area in 1990

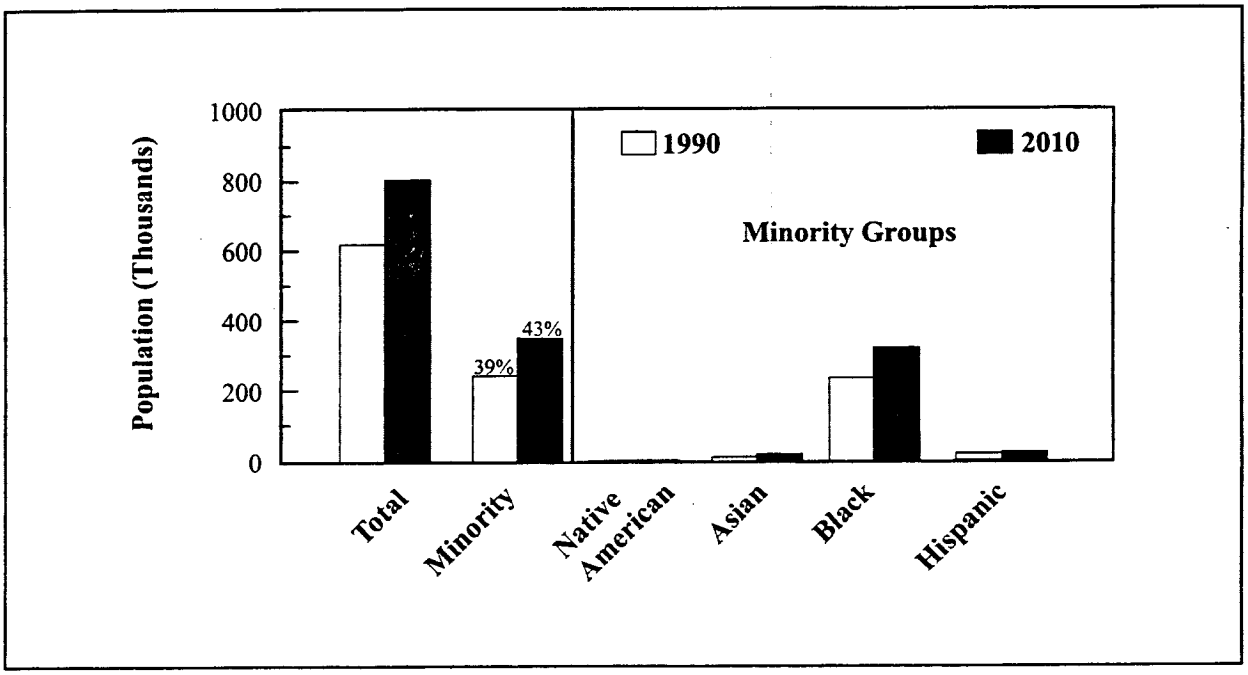


Figure H-7 Racial and Ethnic Composition of the Minority Population Residing Within 80 Kilometers (50 Miles) of the SRS L-Area in 2010

H.5.4 Environmental Impacts at the Sites

Environmental effects that would result from implementation of the various alternatives are discussed in Chapter 4 and summarized in Table 2-4 of Section 2. It was demonstrated in Chapter 4 that implementation of the alternatives would pose no significant radiological or nonradiological risks to the public. **Table H-4** summarizes the radiological impacts described in Chapter 4 that are relevant to the evaluation of environmental justice. Columns 3 and 4 of the table show the estimated likelihood of latent cancer fatalities for the maximally exposed offsite individual and the surrounding population, respectively, under normal operations over the lifetime of the project. For most of the alternatives, the risk of a latent cancer fatality calculated for the maximally exposed offsite individual was too small to be physically observable. Estimated latent cancer fatalities from accidents were less than those expected for normal operations. As indicated in columns 3 and 4 of the table, as well as the discussions of Chapter 4, implementation of the alternatives would pose no significant radiological risks to the general public, and these risks are independent of the racial, ethnic, and economic composition of potentially affected populations. Thus, implementation of the alternatives would pose no disproportionate risks to minority populations or low-income populations within the general population.

Table H–4 Summary of Radiological Effects of the Alternatives on the Public

Alternative	*Project Duration (years)*	*Estimated Likelihood of an Latent Cancer Fatality for the Offsite Maximally Exposed Individual*	*Estimated Likelihood of an Latent Cancer Fatality Among the Population at Risk*
No Action	35	Essentially Zero	1 in 154,000
1	13	Essentially Zero	1 in 125,000
2	09	Essentially Zero	1 in 118,000
3	09	Essentially Zero	1 in 100,000
4	12	Essentially Zero	1 in 118,000
5	09	Essentially Zero	1 in 91,000
6	10	Essentially Zero	1 in 83,000

H.6 Results for Transportation Routes

As discussed in Chapter 4 of this EIS, no significant radiological or nonradiological risks along representative transportation routes would result from implementation of the alternatives for the treatment and management of sodium-bonded spent nuclear fuel. Therefore, implementation of these alternatives would pose no disproportionately high and adverse risks to minority and low-income groups within the general public.

H.7 Other Environmental Impacts

No significant adverse impacts to biotic resources, air resources, socioeconomics, land use, or cultural resources were identified in Chapter 4. Therefore, no disproportionately high and adverse impacts were identified for any segment of the population. None of the alternatives would have a significant adverse impact on the previously mentioned resources because, under all of the alternatives, all activities associated with the treatment and management of sodium-bonded spent nuclear fuel would take place within existing facilities at ANL-W and SRS.

H.8 CUMULATIVE IMPACTS

Based on the analysis of the environmental impacts evaluated in this EIS, along with the impacts of other past, present, and reasonably foreseeable future activities, no reasonably foreseeable cumulative disproportionate and adverse impacts are expected to affect the surrounding minority and low-income populations.

H.9 REFERENCES

CEQ (Council on Environmental Quality), 1997, *Environmental Justice Guidance Under the National Environmental Policy Act*, Executive Office of the President, Washington, DC, December 10.

Campbell, Paul R., 1996, *Population Projections for States by Age, Sex, Race, and Hispanic Origin: 1995 to 2025* (available at http://www.census.gov/population/www/projections/ppl47. html), U.S. Bureau of the Census, Population Division, October.

DOC (U.S. Department of Commerce), 1992, *1990 Census of Population and Housing, Summary Tape File 3 on CD-ROM*, Bureau of the Census, Washington, DC, May.

DOC (U.S. Department of Commerce), 1999, (available at http://www.census.gov/population/www/projections/stproj.html), U.S. Bureau of the Census.

Appendix I

Ecological Resources

APPENDIX I
ECOLOGICAL RESOURCES

I.1 INTRODUCTION

Table I–1 contains a listing of the scientific names of animal and plant species found in the text. Species are grouped and listed in alphabetical order by common name.

Table I–1 Scientific Names of Animal and Plant Species Referred to in the Text

Common Name	*Scientific Name*
Mammals	
Black-tailed jackrabbit	*Lepus californicus*
Coyote	*Canis latrans*
Eastern cottontail	*Sylvilagus floridanus*
Elk	*Cervus elaphus*
Feral hog	*Sus scrofa*
Gray fox	*Urocyon cinereoargenteus*
Gray wolf	*Canis lupus*
Mountain lion	*Felis concolor*
Mule deer	*Odocoileus hemionus*
Pronghorn	*Antilocapra americana*
Pigmy rabbit	*Brachylagus idahoensis*
Raccoon	*Procyon lotor*
Townsend's big-eared bat	*Corynorhinus townsendii*
Townsend's ground squirrel	*Spermophilus townsendii*
Whitetail deer	*Odocoileus virginianus*
Birds	
Bald eagle	*Haliaeetus leucocephalus*
Black vulture	*Coragyps atratus*
Carolina chickadee	*Parus carolinensis*
Common crow	*Corvus brachyrhynchos*
Cooper's hawk	*Accipiter cooperii*
Golden eagle	*Aquila chrysaetos*
Loggerhead shrike	*Lanius ludovicianus*
Peregrine falcon	*Falco peregrinus*
Prairie falcon	*Falco mexicanus*
Red-cockaded woodpecker	*Picoides borealis*
Sage grouse	*Centrocercus urophasianus*
Sage sparrow	*Amphispiza belli*
Wood stork	*Mycteria americana*
Reptiles	
American crocodile	*Crocodylus acutus*
American alligator	*Alligator mississippiensis*
Eastern box turtle	*Terrapene carolina*
Gopher snake	*Pituophis melanoleucus*
Short-horned lizard	*Phrynosoma douglassi*

Common Name	Scientific Name
Amphibians	
Slimy salamander	*Plethodon glutinosus*
Fish	
American shad	*Alosa sapidissima*
Black crappie	*Pomoxis nigromaculatus*
Blueback herring	*Alosa aestivalis*
Bluegill	*Lepomis macrochirus*
Brook trout	*Salvelinus fontinalis*
Creek chubsucker	*Erimyzon oblongus*
Dusky shiner	*Notropis cummingsae*
Kokanee salmon	*Oncorhynchus nerka*
Lake chubsucker	*Erimyzon sucetta*
Largemouth bass	*Micropterus salmoides*
Mosquitofish	*Gambusia affinis*
Mountain whitefish	*Prosopium williamsoni*
Mud sunfish	*Acantharchus pomotis*
Rainbow trout	*Oncorhynchus mykiss*
Redbreast sunfish	*Lepomis auritus*
Redfin pickerel	*Esox americanus*
Shorthead sculpin	*Cottus confusus*
Speckled dace	*Rhinichthys osculus*
Spotted sunfish	*Lepomis punctatus*
Striped bass	*Morone saxatilis*
Sunfish	*Lepomis spp.*
Threadfin shad	*Dorosoma petenense*
Yellow bullhead	*Ictalurus natalis*
Yellowfin shiner	*Notropis lutipinnis*
Mollusks	
Giant oyster	*Crassostrea gigantissima*
Plants	
American ginseng	*Panax quinquefolium*
Bald cypress	*Taxodium distichum*
Big sagebrush	*Artemisia tridentata*
Bluebunch wheatgrass	*Agropyron spicatum*
Bottlebrush squirreltail	*Sitanion hystrix*
Button snakeroot	*Erynglum yuccifolium*
Cottonwood	*Populus spp.*
Crested wheatgrass	*Agropyron desertorum*
Cypress	*Taxodium spp.*
Giant wildrye	*Elymus condensatus*
Gray horsebrush	*Tetradymia canescens*
Green rabbitbrush	*Chrysothamnus greenei*
Hickory	*Carya spp.*
Indian ricegrass	*Oryzopsis hymenoides*
Juniper	*Juniperus spp.*
Loblolly pine	*Pinus taeda*
Longleaf pine	*Pinus palustris*
Low sagebrush	*Artemisia arbuscula*
Needle-and-tread grass	*Stipa comata*
Oak	*Quercus spp.*
Oconee azalea	*Rhododendron flammeum*

Common Name	Scientific Name
Pine	*Pinus spp.*
Poverty-weed	*Monolepis nuttaliana*
Prickly pear cactus	*Opuntia spp.*
Rabbitbrush	*Chrysothamnus spp.*
Redroot	*Lachnanthese carolinianum*
Rush	*Juncus spp.*
Sagebrush	*Artemisia spp.*
Saltbush	*Atriplex spp.*
Slash pine	*Pinus elliottii*
Smooth purple coneflower	*Echinacea laevigata*
Thickspike wheatgrass	*Agropyron dasytachyum*
Threetip sagebrush	*Artemisia tripartita*
Tupelo	*Nyssa slyvotica*
Utah juniper	*Juniperus osteosperma*
Western wheatgrass	*Agropyron smithii*
Willow	*Salix spp.*
Winterfat	*Eurotia lanata*

Appendix J

Federal Register Notices

DEPARTMENT OF ENERGY

Notice of Intent To Prepare an Environmental Impact Statement for Electrometallurgical Treatment of Sodium-Bonded Spent Nuclear Fuel in the Fuel Conditioning Facility at Argonne National Laboratory-West, Idaho National Engineering and Environmental Laboratory, Idaho

AGENCY: U.S. Department of Energy.

ACTION: Notice of intent to prepare an environmental impact statement.

SUMMARY: The Department of Energy announces its intent to prepare an Environmental Impact Statement (EIS) pursuant to the National Environmental Policy Act (NEPA) for the proposed electrometallurgical treatment of Department of Energy-owned sodium-bonded spent nuclear fuel in the Fuel Conditioning Facility at Argonne National Laboratory-West (ANL–W). ANL–W, a center of nuclear technology development and testing, is located on the Idaho National Engineering and Environmental Laboratory (INEEL) site in southeastern Idaho. The Department proposes to treat its inventory of sodium-bonded spent nuclear fuel to remove and stabilize the reactive metallic sodium constituent and to produce metal and ceramic waste forms, considered to be high-level waste, that would facilitate interim storage and ultimate disposal of this material. The EIS will evaluate reasonable action alternatives to electrometallurgical treatment in the Fuel Conditioning Facility at ANL-W and a no-action alternative. The Department invites the general public, other Federal agencies, American Indian tribes, state and local governments, and all other interested

parties to comment on the scope of this EIS.

DATES: To ensure consideration in the preparation of the draft EIS, comments should be transmitted or postmarked by April 8, 1999. Comments submitted after that date will be considered to the extent practicable.

The Department will conduct public scoping meetings in Idaho Falls and Boise in Idaho, near the Department's Savannah River Site (SRS) in South Carolina, and in the Washington, DC area, to provide the public with information about the proposed project and to receive oral and written comments on the scope of the EIS, including reasonable alternatives and environmental issues that the Department should consider. The dates, times, and locations for these public meetings are as follows:

March 9, 1999 (6:00 pm–9:00 pm)
Shilo Inn, 780 Lindsay Blvd., Idaho Falls, ID 83402, (208) 523–0088

March 11, 1999 (6:00 pm–9:00 pm)
Boise Centre on the Grove, 850 West Front Street, Boise, ID 83702, (208) 336–8900

March 15, 1999 (6:00 pm–9:00 pm)
North Augusta Community Center, 495 Brookside Avenue, North Augusta, SC 29842, (803) 441–4290

March 18, 1999 (2:00 pm–5:00 pm)
Hyatt Regency Crystal City, 2799 Jefferson Davis Highway, Arlington, VA 22202, (703) 418–1234

These public scoping meetings will also be announced in local media at least 15 days prior to the meeting dates. During the first hour of each meeting attendees may register, view displays and discuss issues and concerns informally with Department representatives, after which there will be a formal presentation, a follow-on question, answer, and comment period, and the opportunity for additional informal discussions.

ADDRESSES: Written comments on the scope of the EIS, requests to speak at the public scoping meetings, requests for special arrangements to enable participation at scoping meetings (e.g., an interpreter for the hearing impaired), requests to be placed on the EIS document distribution list, and questions concerning the project should be sent to: Susan Lesica, Document Manager, Office of Nuclear Facilities Management, Office of Nuclear Energy, Science, and Technology, U.S. Department of Energy, NE–40, 19901 Germantown Road, Germantown, Maryland 20874–1290

Interested parties may also submit comments and requests by facsimile to (877) 621–8288, or they may call (877) 450–6904 to leave a detailed message with their comments and requests. These are both toll-free telephone numbers. Comments and requests may also be submitted by electronic mail to emtEIS@hq.doe.gov.

FOR FURTHER INFORMATION CONTACT: For general information on the Department of Energy NEPA process, please contact: Carol Borgstrom, Director, Office of NEPA Policy and Assistance, Office of Environment, Safety and Health, U.S. Department of Energy, EH–42, 1000 Independence Avenue, SW, Washington, DC 20585–0119, 202–586–4600 or leave a message at 1–800–472–2756.

SUPPLEMENTARY INFORMATION:

Background

The Department of Energy is responsible for the safe and efficient management of 250 different types of spent nuclear fuel, including its ultimate disposition (which is expected to be disposal in a geologic repository). Some Department spent fuels may be suitable for disposal with little or no stabilizing treatment. Other spent fuel types may not be suitable for disposal without significant treatment or stabilization.

One type of spent nuclear fuel that may not be suitable for disposal without treatment is sodium-bonded spent nuclear fuel. Sodium-bonded spent nuclear fuel contains metallic sodium, a highly reactive material. Metallic sodium reacts vigorously with water or moist air producing heat, potentially explosive hydrogen gas, and sodium hydroxide, a corrosive substance. Sodium metal was used as a heat transfer medium within the stainless steel cladding of sodium-bonded fuel and as coolant in the nuclear reactors in which these fuels were used. To the extent possible, the highly reactive sodium has been removed from external surfaces of these fuels after their use, but a portion remains bonded to the uranium metal alloy fuel within the cladding and cannot be removed without further treatment. The presence of reactive or pyrophoric material, such as metallic sodium, could complicate the process of qualifying and licensing such spent fuel for disposal, which would require data and predictive analyses sufficient to demonstrate that emplacement of the spent fuel would not adversely affect a repository's ability to protect the environment and public health.

The Department believes that treatment to remove metallic sodium and convert this spent nuclear fuel into a compact waste form would reduce complications of disposal qualification and licensing. Technologies for spent nuclear fuel treatment that might facilitate such qualification and licensing should therefore be considered in reaching a disposition decision for Department-owned sodium-bonded fuels. One such technology for sodium-bonded spent fuel disposition is the electrometallurgical treatment technique that the Department is developing and demonstrating at the Argonne National Laboratory. This technology is currently the most developed for treatment of sodium-bonded spent fuel. In addition to electrometallurgical treatment, the Department will examine all reasonable alternative technologies and assess the technical risks associated with these various potential solutions.

In a 1995 report, the National Research Council Committee on Electrometallurgical Techniques for DOE Spent Fuel Treatment recommended that the Department confirm the technical feasibility and cost effectiveness of electrometallurgical treatment of its sodium-bonded spent nuclear fuel through a technology demonstration using sodium-bonded spent nuclear fuel that had been removed from the Experimental Breeder Reactor–II (EBR–II) at ANL–W. Prior to acting on the recommendation, the Department prepared the Environmental Assessment for the Electrometallurgical Treatment Research and Demonstration Project in the Fuel Conditioning Facility at Argonne National Laboratory-West (DOE/EA–1148) and issued a Finding of No Significant Impact on May 15, 1996. The demonstration project addresses both kinds of spent fuel assemblies in the EBR–II spent nuclear fuel inventory. These are driver fuel assemblies and blanket fuel assemblies, and they total about 26 metric tons of heavy metal (MTHM).

One MTHM is equal to 2,200 pounds of uranium, thorium, or plutonium. The driver fuel contains highly enriched uranium and was used in the active region of the nuclear reactor core. Blanket fuel contains depleted uranium and was used in areas around and near the driver fuel in the reactor core. The demonstration project now nearing completion involves treatment of 100 EBR–II driver assemblies and 25 EBR–II blanket assemblies (approximately 1.6 MTHM, or only 6.25% of the EBR–II inventory) in the Fuel Conditioning Facility at ANL–W. The research and demonstration project was initiated in June 1996 and is scheduled to be completed in August 1999.

The National Research Council is continuing to evaluate the electrometallurgical treatment research

and demonstration project. In its most recent report titled, Electrometallurgical Techniques for U.S. Department of Energy Spent Fuel Treatment—Spring 1998 Status Report on Argonne National Laboratory's R&D Activity (National Academy Press, Washington, DC, 1998), the Council acknowledged progress in the demonstration and recommended that the demonstration be carried to completion. The Department believes that this progress and the absence of significant roadblocks to successful completion of the demonstration warrant proposing electrometallurgical treatment of the remainder of the EBR–II and other sodium-bonded spent fuels (i.e., a total of 62 MTHM) and is initiating the environmental review process under NEPA. Accordingly, the Department is announcing its intent to prepare an EIS for the proposed treatment of the remainder of Department sodium-bonded spent nuclear fuel.

Data from the ongoing demonstration project will be used in preparing the EIS. The National Research Council will issue a final report on the technology demonstration upon completion of the demonstration project. DOE will consider the Council's report in reaching a decision regarding the disposition of sodium-bonded spent nuclear fuel.

Purpose and Need for Agency Action

In a 1995 agreement with the State of Idaho [Settlement Agreement and Consent Order issued by the Court on October 17, 1995, in the actions *Public Service Co. of Colorado* v. *Batt*, No. CV 91–0035–S–EJL (D. Id.), and *United States* v. *Batt*, No. CV 91–0054–EJL (D. Id.), the Department committed to remove all spent nuclear fuel from Idaho by 2035. More than 98 percent of the Department's sodium-bonded spent nuclear fuel is located at INEEL near Idaho Falls, Idaho, and is subject to the requirements of the Settlement Agreement and Consent Order. The remaining Department sodium-bonded spent nuclear fuel included in the proposed action is at the Hanford Reservation in Richland, Washington, the Sandia National Laboratories in Albuquerque, New Mexico, and the Oak Ridge National Laboratory in Oak Ridge, Tennessee. In order to remove sodium-bonded spent nuclear fuel from the State of Idaho to meet the terms of the Settlement Agreement and Consent Order referenced above, the Department believes the best approach would be to stabilize or remove the reactive metallic sodium constituent and prepare a waste form that may be more assuredly demonstrated to be acceptable for disposal.

It is also prudent to evaluate the electrometallurgical treatment proposal and alternative technologies now, while the Department is performing site characterization activities for a potential geologic repository. Contemplated waste forms resulting from treatment or packaging of sodium-bonded spent fuel should be developed as much as possible in parallel with any repository development to promote consistency between the two efforts and to minimize technical risks associated with waste form qualification and acceptance for geologic disposal. While the alternative technologies for treatment of sodium-bonded spent fuel may not be as mature as the electrometallurgical treatment technology, their potential utility can be assessed in this EIS. Should the Department decide, after completing this EIS, to pursue a disposition path other than electrometallurgical treatment, there will still be sufficient time to develop an alternative technology. If a treatment technology decision is significantly delayed, however, the Department could functionally lose its expertise and corporate experience in the specialized electrometallurgical treatment technology at ANL–West, which would hamper future consideration and increase the cost of electrometallurgical treatment for sodium-bonded spent fuel disposal. Therefore, the Department believes it is prudent to proceed now with this EIS for electrometallurgical treatment of sodium-bonded spent fuel.

Proposed Action

The Department proposes to treat its sodium-bonded spent nuclear fuel [1] using the electrometallurgical treatment process in the Fuel Conditioning Facility at ANL–W. Electrometallurgical processing involves the dissolution of spent nuclear fuel by use of an electric current in a molten salt mixture. The uranium in the fuel would be collected from a molten salt mixture at the cathode and subsequently melted and cast into metal ingots. The metal cladding from the fuel elements and noble metal fission products would be retrieved undissolved from the anode, melted, and cast into metal ingots. Remaining fission products and all transuranic elements would be removed from the molten salt mixture by ion exchange and subsequently isolated in a ceramic waste form. In this process, the metallic sodium in the spent nuclear fuel would be converted to non-reactive sodium chloride (same composition as table salt) and incorporated in the ceramic waste form.

Based on available information, the Department believes the electrometallurgical treatment process would produce metal and ceramic high-level radioactive waste forms that could be qualified and licensed for disposal. In addition, uranium would be separated from both the driver fuel and the blanket fuel and not disposed of. The highly enriched uranium separated from the driver fuel assemblies would be immediately blended down in the Fuel Conditioning Facility to form low-enriched uranium. This low-enriched uranium and the depleted uranium that would be separated from blanket fuel assemblies would be cast as metal ingots and stored with other uranium metal inventories at INEEL. The disposition of these materials would be included in future Departmental decisions regarding other similar materials.

The sodium-bonded spent nuclear fuel inventory being proposed for electrometallurgical treatment totals approximately 62 MTHM. This inventory of sodium-bonded spent nuclear fuel is currently stored as follows:

- Approximately 24 MTHM of EBR–II sodium-bonded driver and blanket assemblies currently stored at ANL–W and approximately 2 MTHM at the Idaho Nuclear Technology and Engineering Center (INTEC), both located at INEEL.
- Approximately 35 MTHM of sodium-bonded spent nuclear fuel from the Fermi-1 reactor, currently stored at INTEC.
- Less than one MTHM consisting of six irradiated sodium-bonded fuel assemblies and a number of sodium-bonded spent nuclear fuel pins currently stored at the Hanford Reservation near Richland, Washington.
- Less than 0.1 MTHM consisting of experimental capsules currently stored at INTEC and Clinch River Breeder Reactor Program experimental capsules currently stored at Sandia National Laboratories, Albuquerque, New Mexico.
- Less than 0.01 MTHM consisting of miscellaneous fast reactor development fuel currently stored at Oak Ridge National Laboratory, Oak Ridge, Tennessee.

[1] The Department has no plan or intention to apply this technology to any other types of spent nuclear fuel. Nevertheless, the Department can foresee a potential need to treat small quantities of certain spent fuels if a non-treatment (e.g., high integrity can) approach to disposing of such spent fuels were to be determined not to meet disposal requirements. In that case, electrometallurgical treatment might be among the reasonable alternative treatment technologies that would be considered.

The sodium-bonded spent nuclear fuels located at the Hanford Reservation, Oak Ridge, and Sandia can be transported to INEEL pursuant to the Record of Decision (60 FR 28680, June 1, 1995) for the Department of Energy's Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Final Environmental Impact Statement (DOE/EIS–0203–F), under the Settlement Agreement and Consent Order described above. These spent fuels pose the same waste form acceptability issues and are amenable to the same treatments as the EBR–II and Fermi-1 fuels stored at INEEL.

Alternatives To Be Evaluated

The Department has identified the following alternatives to the proposed electrometallurgical treatment of sodium-bonded spent nuclear fuel in the Fuel Conditioning Facility at ANL–W.

A. *No Action Alternative:* Under this alternative, the Department would not treat its sodium-bonded spent nuclear fuel to facilitate disposal. Analyses will address the viability of disposal without treatment, and the impacts of continued storage at current locations. Both temporary storage (to await alternative technology development) and indefinite storage (in lieu of disposal) will be considered in these analyses. Indefinite storage of spent nuclear fuel in Idaho would not be consistent with the Settlement Agreement and Consent Order in which the Department committed to remove all spent nuclear fuel from Idaho by 2035.

B. *Technology Alternatives:* The National Research Council independently assessed other treatment technologies as possible alternatives to electrometallurgical treatment for EBR–II sodium-bonded spent nuclear fuel. It concluded that all of the alternative treatment processes evaluated, except the Plutonium-Uranium Extraction (PUREX) process, are at an early stage of development. Significant research, development, and demonstrations would be required to develop these alternative treatment processes to the level of technical maturity of the electrometallurgical treatment process for sodium-bonded spent fuel. However, the Department will examine and analyze these alternative technologies:

1. PUREX Process. This solvent extraction method for separating and purifying uranium, plutonium, and other radionuclides from spent nuclear fuel and irradiated targets is presently practiced at the SRS for stabilization of materials that are not suitable for prolonged storage in their present forms, and as such pose potential health and safety risks. In the Savannah River Site Spent Nuclear Fuel Management EIS, the Department is currently evaluating use of the PUREX process for stabilizing approximately 17 MTHM of previously declad EBR–II spent nuclear fuel stored at the SRS site. Use of the PUREX facility to treat sodium-bonded spent nuclear fuel being considered under this alternative would require development of specific processes for removing the stainless-steel cladding and sodium from the spent fuel.

The Department intends to evaluate the PUREX process at SRS as an alternative to electrometallurgical treatment of the sodium-bonded spent fuel inventory. Material streams from the PUREX process would be uranium trioxide, plutonium metal, high-level waste in the form of borosilicate glass canisters, and grouted low-level waste.

2. High-Integrity Cans. Under this alternative, the spent fuel would be placed in high-integrity cans, after as little treatment as necessary, to prepare it for disposal. This alternative would include removal of as much of the metallic sodium as possible from the spent fuel prior to loading it in the cans.

3. Glass Material Oxidation and Dissolution System (GMODS). The basic concept is to combine unprocessed sodium-bonded spent nuclear fuel and a sacrificial oxide, lead-borate glass, in a glass melter at a temperature of 800–1000 °C. The uranium and the plutonium in the spent fuel would be converted into oxides and dissolved in the glass. Options to be analyzed are direct production of a borosilicate glass waste form from the melt, using the melt as a feed to the PUREX process, and coupling GMODS to the SRS Defense Waste Processing Facility, where the melt would be fed directly to the existing glass melter. Due to the powerful dissolution and oxidation properties of the lead-borate glass melt, containment is a concern, and a water-cooled, cold-wall, induction-heated melter must be used.

4. Melt and Dilute Process. The process would be similar to that proposed for the treatment of aluminum-based spent nuclear fuels at the SRS. The sodium-bonded spent fuel would be chopped and melted at approximately 650 to 850 °C and then diluted by the addition of depleted uranium and iron.

5. Chloride Volatility Process. This process would use the differences in volatilities of chloride compounds to separate the constituents of spent nuclear fuel. The major steps are: (1) high-temperature chlorination at about 1500 °C and conversion of metallic fuel and cladding to gaseous chloride compounds; (2) removal of the transuranic chlorides and most of the fission products in a molten zinc chloride bed at approximately 400 °C; (3) condensation of the other chlorides (e.g., uranium hexachloride) in a series of fluidized beds and condensers at successively lower temperatures; and (4) zinc chloride regeneration/recycling. The transuranics and fission product chlorides would then be converted into either fluorides or oxides for disposal.

6. Direct Plasma Arc-Vitreous Ceramic Process. In this process, the spent nuclear fuel would be melted and oxidized with the help of an oxygen lance in a rotating furnace containing molten ceramic materials at a temperature of 1600 °C or higher. A direct current plasma torch would supply the energy required in the process. Rotation would be used to keep the molten pool in the furnace. When the spent fuel is homogeneously melted and oxidized throughout the ceramic, rotation would be slowed to allow the molten vitreous ceramic to pour out by gravity flow into a canister.

C. *Location Alternatives:* An alternative location for electrometallurgical treatment on the INEEL site is the Test Area North Hot Cell Facility. This alternative to the Fuel Conditioning Facility at ANL–W will be evaluated in the EIS.

U.S. Nonproliferation Policy Implications

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. Consistent with this policy, the proposed action would not separate plutonium from the processed sodium-bonded spent fuels. Further, by removing and diluting the highly enriched uranium in the sodium-bonded driver fuel to low-enriched uranium, the proposed project would support the U.S. goal of minimizing civilian use of highly enriched uranium. However, to address the concerns that the treatment of this fuel could encourage reprocessing in other countries, the Department (Office of Nonproliferation and National Security) will assess the nonproliferation impacts of all the treatment technologies in the draft EIS. This assessment will be made publicly available during the EIS process. The combination of the information contained in the draft EIS, the public comment in response to the draft EIS, and the nonproliferation impacts assessment report will enable

the Department to make a sound decision regarding how to manage the sodium-bonded spent nuclear fuel.

Preliminary Identification of Environmental Issues

The issues listed below have been tentatively identified for analysis in the EIS. This list is presented to facilitate public comment on the scope of the EIS. It is not intended to be all-inclusive or to predetermine the potential impacts of any of the alternatives. The Department seeks public comment on the adequacy and inclusiveness of the following issues.

- Potential impact on ecosystems, including air quality, surface, and groundwater quality, and plants and animals.
- Potential health and safety impact to on-site workers and to the public resulting from operations, including reasonably foreseeable accidents.
- Potential health and safety, environmental, and other impact related to the transport of spent nuclear fuel for treatment.
- Considerations related to the generation, treatment, storage, and disposal of wastes, including the potential acceptability of waste forms at a geologic repository.
- Potential cumulative impacts of electrometallurgical and alternative treatment process operations, including relevant impact from other past, present, and reasonably foreseeable activities at the operation site.
- Potential impact on cultural resources.
- Potential socioeconomic impact, including any disproportionate impacts on minority and low income populations.
- Pollution prevention and waste minimization opportunities.

Related NEPA Documentation

NEPA documents that have been or are being prepared for activities related to the proposed action include, but are not limited to, the following:

- U.S. Department of Energy, "Electrometallurgical Treatment Research and Demonstration Project in the Fuel Conditioning Facility at Argonne National Laboratory-West; Environmental Assessment," DOE/EA–1148, May 1996
- U.S. Department of Energy, "Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management; Final Environmental Impact Statement," DOE/EIS–0203–F, April 1995, and Record of Decision, May 30, 1995
- U.S. Department of Energy, "Savannah River Site, Spent Nuclear Fuel Management, Draft Environmental Impact Statement," DOE/EIS–0279D, December 1998
- U.S. Department of Energy, "Draft Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada," DOE/EIS–0250—in preparation

Public Involvement Opportunities

The Department encourages public involvement in the preparation of the EIS and solicits public comments on its scope and content, as well as public participation at the public scoping meetings in Idaho, South Carolina, and the Washington, DC area. Department of Energy personnel will be available at the scoping meetings to explain the proposed project and answer questions. The Department will designate a facilitator for the scoping meetings. At the opening of each meeting, the facilitator will establish the order of speakers and will announce any additional procedures necessary for conducting the meeting. Additionally, during the first hour of each meeting attendees may register, view displays and discuss issues and concerns informally with Department representatives, after which there will be a formal presentation, a question and answer, and comment period, and the opportunity for additional informal discussions. To ensure that all persons wishing to make a presentation during the period for questions and answers or comments are given the opportunity to speak, a five-minute limit may be applied for each speaker, except that public officials and representatives of groups would be allotted ten minutes each. The Department encourages those providing oral comments to also submit them in writing. Comment cards will be available at the meetings for those who prefer to submit their comments in writing. Speakers may be asked clarifying questions to ensure that the Department representatives fully understand the comments and suggestions made by meeting participants, but the scoping meetings will not be conducted as evidentiary hearings.

The Department will make transcripts of public scoping meetings, copies of background documents, and other materials related to the proposed project and the development of the EIS available for public review in the following reading rooms:

Washington, DC: U.S. Department of Energy, Freedom of Information Reading Room, Forrestal Building, Room 1E–190, 1000 Independence Avenue, SW, Washington, DC 20585–0117, 202–586–3142

Idaho Falls, Idaho: Idaho National Engineering and Environmental Laboratory, DOE—Idaho Operations Office Public Reading Room, 1776 Science Center Drive, Idaho Falls, ID 83415, 208–526–0271

Richland, Washington: [for vicinity of the Hanford Reservation], DOE Public Reading Room, 2770 University Drive, CIC, Room 101L, Richland, WA 99352, 509–372–7443, (Fax) 509–372–7444

Albuquerque, New Mexico: [for vicinity of Sandia National Laboratories], University of New Mexico, Government Information Department, Zimmerman Library, Albuquerque, NM 87131–1466, 505–277–0582

Aiken, South Carolina: [for vicinity of the Savannah River Site], University of South Carolina—Aiken, Gregg-Graniteville Library, 171 University Parkway, Aiken, SC 29803, 803–648–6851

Oak Ridge, Tennessee: [for vicinity of the Oak Ridge National Laboratory], DOE Public Reading Room, 230 Warehouse Road, Bldg 1916–T–2, Suite 300, Oak Ridge, TN 37831, 423–241–4780 and DOE Information Resource Center, 105 Broadway Avenue, Oak Ridge, TN 37830, 423–241–4582

NEPA Process

The EIS for Electrometallurgical Treatment of Sodium-Bonded Spent Nuclear Fuel in the Fuel Conditioning Facility at ANL–W will be prepared in accordance with the NEPA of 1969, the Council on Environmental Quality's Regulations for Implementing the Procedural Provisions of NEPA (40 CFR Parts 1500–1508), and the U.S. Department of Energy NEPA Implementing Procedures (10 CFR Part 1021).

A 45-day comment period on the draft EIS is planned, during which public hearings to receive comments will be held. The draft EIS is scheduled to be issued in July 1999. Availability of the draft EIS, the dates of the public comment period, and information about the public hearings will be announced in the **Federal Register** and in local news media when the draft EIS is distributed. The final EIS, which will consider and respond to the public comments received on the draft EIS, is scheduled to be issued in December 1999. No sooner than 30 days after the U.S. Environmental Protection Agency's notice of availability of the final EIS is published in the **Federal Register**, the Department will issue its Record of

Decision and publish it in the **Federal Register**.

Signed in Washington, DC, this 16th day of February 1999.

Peter N. Brush,

Principal Deputy Assistant Secretary, Environment, Safety and Health.

[FR Doc. 99–4289 Filed 2–19–99; 8:45 am]

BILLING CODE 6450–01–P

DEPARTMENT OF ENERGY

Availability of the Draft Environmental Impact Statement (EIS) for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel

AGENCY: Department of Energy.

ACTION: Notice of availability.

SUMMARY: The Department of Energy announces the availability of the "Draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel" (DOE/EIS–0306D) for public review and comment. This draft EIS, prepared under the National Environmental Policy Act (NEPA), assesses the potential environmental impacts associated with the treatment and management of the Department's sodium-bonded spent nuclear fuel in one or more spent nuclear fuel management facilities: Argonne National Laboratory-West at the Idaho National Engineering and Environmental Laboratory (near Idaho Falls, Idaho) and either the F-Canyon or Building 105–L at the Savannah River Site (near Aiken, South Carolina). The Department is considering ways to prepare the sodium-bonded spent nuclear fuel for disposal in a geologic repository. Such treatment would remove or stabilize the chemically reactive sodium in the fuel. The EIS evaluates reasonable alternatives for treatment of the Department's sodium-bonded spent nuclear fuel and a no-action alternative. The Department has no preferred alternative at this time.

DATES: The Department invites the general public, other Federal agencies, Tribal, State and local Governments to provide comments on this draft EIS. The 45-day comment period extends through September 13, 1999. To ensure consideration in the preparation of the final EIS, comments should be transmitted or postmarked by September 13, 1999. Comments submitted after that date will be considered to the extent practicable. The information obtained during the comment period will assist the Department in preparing the final EIS, which is scheduled to be completed by December 31, 1999.

During the review period, the Department will hold public hearings to discuss the proposed action and to receive oral and written comments on the draft EIS. The hearings are scheduled for the following dates, times, and locations.

August 17, 1999, (6:00 p.m. to 9:00 p.m.), North Augusta Community Center, 495 Brookside Avenue, North Augusta, South Carolina 29842, (803) 441–4290

August 24, 1999, (6:00 p.m. to 9:00 p.m.), Owyhee Plaza Hotel, 1109 Main Street, Boise, Idaho 83702, (208) 343–4611

August 26, 1999, (6:00 p.m. to 9:00 p.m.), Shilo Inn, 780 Lindsay Boulevard, Idaho Falls, Idaho 83402, (208) 523–0088

August 31, 1999, (2:00 p.m. to 5:00 p.m.), Hyatt Regency Crystal City, 2799 Jefferson Davis Highway, Arlington, Virginia 22202, (703) 418–1234

The format for the hearings will include an opportunity for informal discussions with project personnel before and after the formal presentation. After the presentation, there will be an opportunity to provide comments on the draft EIS to Departmental representatives. The Department encourages those providing oral comments at the hearings to also submit them in writing. Comment forms will be available at the hearings for those who prefer to submit their comments in writing.

The Department will make transcripts of the draft EIS hearings, copies of background documents, and other materials related to the proposed project and development of the EIS available for public review in the following reading rooms:

Washington, D.C.: U. S. Department of Energy, Freedom of Information Reading Room, Forrestal Building, Room 1E–190,

1000 Independence Avenue, S.W., Washington, D.C. 20585–0117, 202–586–3142

Idaho Falls, Idaho: Idaho National Engineering and Environmental Laboratory, DOE-Idaho Operations Office Public Reading Room, 1776 Science Center Drive, Idaho Falls, Idaho 83415, 208–526–0271

Richland, Washington: DOE Public Reading Room, 2770 University Drive, CIC, Room 101L, Richland, Washington 99352, 509–372–7443

Albuquerque, New Mexico: University of New Mexico, Government Information Department, Zimmerman Library, Albuquerque, New Mexico 87131, 505–277–0582

Aiken, South Carolina: University of South Carolina-Gregg Graniteville Library, 471 University Parkway, Aiken, South Carolina 29803, 803–648–6851

Oak Ridge, Tennessee: DOE Public Reading Room, 230 Warehouse Road, Building 1916–T–2, Suite 300, Oak Ridge, Tennessee 37831, 423–241–4780 and the DOE Information Resource Center, 105 Broadway Avenue, Oak Ridge, Tennessee 37830, 423–241–4582

ADDRESSES: Written comments on the draft EIS, requests for special arrangements to enable participation in the hearings (e.g., an interpreter for the hearing impaired), requests to be placed on the EIS distribution list, and questions concerning the project should be sent to: Ms. Susan M. Lesica, EIS Document Manager, Office of Nuclear Facilities Management, Office of Nuclear Energy, Science and Technology, U.S. Department of Energy, NE–40, 19901 Germantown Road, Germantown, Maryland 20874–1290.

Comments and requests may also be submitted by toll free facsimile to (877) 621–8288 or telephone (877) 450–6904. Comments and requests may also be submitted by electronic mail to emtEIS@hq.doe.gov.

FOR FURTHER INFORMATION CONTACT: For general information on the Department's National Environmental Policy Act (NEPA) process, please contact: Ms. Carol M. Borgstrom, Director, Office of NEPA Policy and Assistance (EH–42), Office of Environment, Safety and Health, U.S. Department of Energy, 1000 Independence Avenue, SW Washington, DC 20585–0119; or telephone (202) 586–4600 or leave a message at 1–800–472–2756.

SUPPLEMENTARY INFORMATION: The Department is responsible for the safe and efficient management of several types of spent nuclear fuel including its ultimate disposition (that is expected to be disposal in a geologic repository). Some Departmental spent fuels may be suitable for disposal with little or no stabilizing treatment. Other spent fuel types may not be suitable for disposal without significant treatment or stabilization.

One type of spent nuclear fuel that may not be suitable for disposal without treatment is sodium-bonded spent nuclear fuel. Sodium-bonded spent nuclear fuel contains metallic sodium, a highly chemically reactive material. Metallic sodium reacts vigorously with water or moist air producing heat, potentially explosive hydrogen gas, and sodium hydroxide, a corrosive substance. Sodium metal was used as a heat transfer medium within the stainless steel cladding of sodium-bonded fuel and as coolant in the nuclear reactors in which these fuels were used. To the extent possible, the highly reactive sodium has been removed from external surfaces of these fuels after their use, but a portion remains bonded to the uranium metal alloy fuel within the cladding and cannot be removed without further treatment. The presence of reactive or pyrophoric material such as metallic sodium, could complicate the process of qualifying and licensing such spent fuel for disposal, which would require data and predictive analyses sufficient to demonstrate that emplacement of the spent fuel would not adversely affect a repository's ability to protect the environment and public health.

The Department believes that treatment to remove metallic sodium and convert this spent nuclear fuel into a compact waste form would facilitate disposal qualification of this fuel. Technologies for spent nuclear fuel treatment that might facilitate such qualification should, therefore, be considered in deciding how to manage Department-owned sodium-bonded fuels. The EIS analyzes, under the proposed action, six reasonable alternatives that employ one or more of the following technology options: electrometallurgical treatment; the plutonium-uranium extraction process; packaging in high-integrity cans; and the melt and dilute treatment process. The EIS also evaluates a no action alternative, under which the sodium-bonded spent nuclear fuel would continue to be stored indefinitely; The Department would pursue research and development of a new treatment technology or would directly dispose of the sodium-bonded spent nuclear fuel in high-integrity cans without treatment.

The Department has not identified a preferred alternative in the draft EIS. Environmental analysis in this EIS, public comments, the findings of an independent cost study and a nonproliferation report that are being prepared concurrently with the EIS, as well as other program policy factors, will be considered in determining a preferred alternative in the final EIS. A Record of Decision will be issued no sooner than 30 days after the final EIS has been distributed.

Issued in Washington, D.C., this 22, day of July 1999.

William D. Magwood, IV,

Director, Office of Nuclear Energy, Science and Technology.

[FR Doc. 99–19522 Filed 7–29–99; 8:45 am]

BILLING CODE 6450–01–P

DEPARTMENT OF ENERGY

Draft Environmental Impact Statement (EIS) for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel; Public Comment Period Extension

AGENCY: Department of Energy (DOE).

ACTION: Extension of public comment period.

SUMMARY: In response to requests from the public, DOE has decided to extend the deadline for the transmittal of public comments on the "Draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel" (DOE/EIS–0306D) from September 13, 1999, to September 28, 1999.

DATES: Comments should be transmitted or postmarked by September 28, 1999, to ensure consideration. Comments submitted after that date will be considered to the extent practicable.

ADDRESSES: Written comments on the draft EIS, requests to be placed on the EIS distribution list, and questions concerning the project should be sent to: Ms. Susan M. Lesica, EIS Document Manager, Office of Nuclear Facilities Management, Office of Nuclear Energy, Science and Technology, U.S. Department of Energy, NE–40, 19901 Germantown Road, Germantown, Maryland 20874–1290.

Comments and requests may also be submitted by toll-free facsimile to (877) 621–8288 or telephone (877) 450–6904. Comments and requests may also be submitted by electronic mail to sodium.fuel.eis@hq.doe.gov.

FOR FURTHER INFORMATION CONTACT: For general information on DOE's NEPA process, please contact: Ms. Carol M. Borgstrom, Director, Office of NEPA Policy and Assistance (EH–42), Office of Environment, Safety and Health, U.S. Department of Energy, 1000 Independence Avenue, SW, Washington, DC 20585–0119; or telephone (202) 586–4600 or leave a message at 1–800–472–2756.

SUPPLEMENTARY INFORMATION: On July 30, 1999, DOE published a notice in the Federal Register (64 FR 41404) announcing the availability of the draft EIS. DOE received requests from several

parties to extend the comment period. In response to these requests and to ensure that all interested parties have time to comment, the deadline for transmittal of comments has been extended to September 28, 1999. Comments should be postmarked by September 28, 1999, to ensure consideration.

Issued in Washington, DC, this 3rd day of September 1999.

William D. Magwood, IV,

Director, Office of Nuclear Energy, Science and Technology.

[FR Doc. 99–23567 Filed 9–9–99; 8:45 am]

BILLING CODE 6450–01–P

ENVIRONMENTAL PROTECTION AGENCY

[ER–FRL–6244–8]

Environmental Impact Statements; Notice of Availability

Responsible Agency: Office of Federal Activities, General Information (202) 564–7167 OR (202) 564–7153.

Weekly receipt of Environmental Impact Statements, Filed July 19, 1999 Through July 23, 1999, Pursuant to 40 CFR 1506.9.

EIS No. 990253, FINAL EIS, FHW, MN, MN–TH–14 Corridor Reconstruction, MN–TH–60 to I–35, Funding and COE Section 404 Permit Issuance, Blue Earth, Waseca and Steele Counties, MN, Due: August 30, 1999, Contact: Cheryl Martin (651) 291–6120.

EIS No. 990254, DRAFT EIS, FHW, VA, ADOPTION—Grundy Flood Damage Reduction/Highway Upgrade Project, Implementation, Town of Grundy, Buchanan County, CA, Due: August 30, 1999, Contact: Doris Bush (540) 645–1667.

U.S. Department of Transportation's, Federal Highway Administration (FHW) has adopted the Corps of Engineer's, Air National Guard FEIS #950407 filed 8–30–95. FHW was not a Cooperating Agency for the above final EIS. Recirculation of the document is necessary under Section 1506.3(b) of the Council on Environmental Quality Regulations.

EIS No. 990255, DRAFT EIS, FHW, WV, US–35, Funding and COE Section 404 Permit, Mason and Putnam Counties, WV, Due: September 23, 1999, Contact: David E. Bender (304) 347–5928.

EIS No. 990256, FINAL EIS, UMC, AZ, Yuma Marine Corps Air Station (MCAS), To Improve Ordnance Handling and Storage, Construct a new Combat Aircraft Loading Area (CALA); New Station Ordnance Area and Relocation of MCAS Yuma, AZ, Due: August 30, 1999, Contact: Richard Samrah (520) 341–3163.

EIS No. 990257, DRAFT EIS, AFS, MT, Good Creek Resource Management Project, Implementation, Vegetation Treatments and Other Activities to Restore Watershed, Flathead National Forest, Tally Lake Ranger District, Flathead County, MT, Due: September 13, 1999, Contact: Bryan Donner (406) 863–5408.

EIS No. 990258, FINAL EIS, FHW, PA, Central Bradford County Traffic Improvement Project, Construction US 6 Highway through Towanda Borough and North Towanda Township to US 220, Bradford County, PA, Due: August 30, 1999, Contact: Ronald W. Carmichael, PE (717) 221–3461.

EIS No. 990259, FINAL EIS, DOC, PR, VI, Corals and Reef Associated Plants and Invertebrates, Fishery Management Plan, Amendment I Marine Conservation District (MCD), Exclusive Economic Zone (EEZ), Puerto Islands and U.S. Virgin Islands, PR and VI, Due: August 30, 1999, Contact: William Hogarth (727) 570–5305.

EIS No. 990260, DRAFT EIS, FHW, NY, Albany Shaker Road and Watervliet Shakey Road Improvement Project, Construction and Reconstruction, Funding and COE Section 404 Permit, Town of Colonie, Albany County, NY, Due: September 15, 1999, Contact: Harold J. Brown (518) 431–4137.

EIS No. 990261, DRAFT EIS, USN, Surveillance Towed Array Sensor System (SURTASS) Low Frequency Active (LFA), To Improved Capability to Detect Quieter and Harder-to-Find Foreign Submarines, Implementation, Due: October 28, 1999, Contact: Kim DaPaul (703) 604–1233.

EIS No. 990262, DRAFT EIS, DOE, Sodium-Bonded Spent Nuclear Fuel for the Treatment and Management, Candidate Sites are Argonne National Laboratory-West (ANL–W) Located within the boundaries of the Idaho National Laboratory I and the Savannah River Sites (SRS) F-Area and L Area, SC, Due: September 13, 1999, Contact: Susan M. Lesica (301) 903–8755.

EIS No. 990263, FINAL EIS, FRC, MA, Holyoke Hydroelectric Relicensing Project. ((FERC) Nos. 2004–073 and 11607–000), Construction, Operation and Maintenance, Located on the Connecticut River, Hampshire, Hampden and Franklin Counties, MA, Due: August 30, 1999, Contact: Allan E. Creamer (202) 219–0365.

EIS No. 990264, FINAL EIS, BLM, UT, Grand Staircase-Escalante National Monument Management Plan, Implementation, Cedar City, UT, Due: August 30, 1999, Contact: Jerry Meredith (435) 865–5100.

EIS No. 990265, DRAFT EIS, APH, Fruit Fly Cooperative Control Program, Eradication Program, Implementation, Due: October 12, 1999, Contact: Harold T. Smith (310) 734–8565.

Amended Notices

EIS No. 990162, DRAFT EIS, USN, GU, Surplus Navy Property Identified in the Guam Land Use Plan (GLUP '94) for Disposal and Reuse, Implementation, GU, Due: September 15, 1999, Contact: Gerald Gibbons (808) 471–9338. Published FR 05–21–99—Review Period extended from 7–6–99 to 9–15–99.

Dated: July 27, 1999.

William D. Dickerson,

Director, NEPA Compliance Division, Office of Federal Activities.

[FR Doc. 99–19625 Filed 7–29–99; 8:45 am]

BILLING CODE 6560–50–M

Appendix K

Settlement Agreement and Consent Order With the State of Idaho

APPENDIX C SETTLEMENT AGREEMENT

Public Service Co. of Colorado v. Batt
and United States v. Batt

UNITED STATES COURTS
DISTRICT OF IDAHO

OCT 17 1995

8:34 A.M. REC'D _____
LODGED FILED _____

UNITED STATES COURTS

DISTRICT OF IDAHO

SETTLEMENT AGREEMENT

The State of Idaho, through the Attorney General and Governor Philip E. Batt in his official capacity; the Department of-Energy, through the General Counsel and Assistant Secretary for Environmental Management; and the Department of the Navy, through the General Counsel and Director, Naval Nuclear Propulsion Program, hereby agree on this 16th day of October, 1995, to the following terms and conditions to fully resolve all issues in the actions 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.):

A. DEFINITIONS

For purposes of this Agreement, the following definitions shall apply:

1. The "State" shall mean the State of Idaho and shall include the Governor of the State of Idaho and the Idaho State Attorney General.

2. The "federal parties" means U.S. Department of Energy (DOE) and the U.S. Department of the Navy (the Navy), including any successor agencies.

3. "Treat" shall be defined, as applied to a waste or spent fuel, as any method, technique, or process designed change the physical or chemical character of the waste or fuel to render it less hazardous; safer to transport, store, dispose of; or reduce in volume.

4. "Transuranic waste" shall be defined as set forth in the EIS, Volume 2, Appendix E.

5. "One shipment of spent fuel" shall be defined as the transporting of a single shipping container of spent fuel.

6. "High-level waste" shall be defined as set forth in the EIS, Volume 2, Appendix E.

7. "DOE spent fuel" shall be defined as any spent fuel which DOE has the responsibility for managing with the exception of naval spent fuel and commercial spent fuel which DOE has accepted or will take title to pursuant to the Nuclear Waste Policy Act of 1982, 42 U.S.C. _ 10101 et seq. or comparable statute.

8. "Naval spent fuel" shall be defined as any spent fuel removed from naval reactors as a result of refueling overhauls (refueling) or defueling inactivations (defueling).

9. "Metric ton of spent fuel" shall be defined as a metric ton of heavy metal of spent fuel.

10. "Naval reactors" shall be defined as nuclear reactors used aboard naval warships (submarines, aircraft carriers, or cruisers), naval research or training vessels, or at land-based naval prototype facilities operated by the Naval Nuclear Propulsion Program for the purposes of research, development, or training.

11. "Calendar year" shall be defined as the year beginning on January 1, and ending on December 31.

12. "Mixed Waste" shall be defined as set forth in the EIS, Volume 2, Appendix E.

13. "EIS" shall be defined as the Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Program Final Environmental Impact Statement issued April, 1995.

14. "ROD" shall be defined as the Record of Decision issued by DOE on June 1, 1995, concerning the EIS.

15. "INEL" shall be defined as the Idaho National Engineering Laboratory. .

16. "Running Average" shall mean the total number of shipments of naval spent fuel to INEL, or transuranic waste from INEL, over any period' of three years, divided by three.

17. "The Court" shall mean the United States District Court for the District of Idaho before which is pending Public Service Company of Colorado v. Batt, No. CV 91-0035-S-EJL and United States v. Batt, No. CV 91-0054-S-EJL, and any appellate court to which an appeal may be taken, or with which an application for a writ of certiorari may be filed, under applicable law.

B. TRANSURANIC WASTE SHIPMENTS LEAVING IDAHO

1. "DOE shall ship all transuranic waste now located at INEL, currently estimated at 65,000 cubic-meters in volume, to the Waste Isolation Pilot Plant (WIPP) or other such facility designated by DOE, by a target date of December 31, 2015, and in. no event later than December 31, 2018. DOE shall meet the following interim deadlines:

a. The first shipments of transuranic waste from INEL to WIPP or other such facility designated by DOE shall begin by April 30, 1999. ,

b. By December 31, 2002, no fewer than 3,100 cubic meters (15,000 drum-equivalents) of transuranic waste shall have been shipped out of the State of Idaho.

c. After January 1, 2003, a running average of no fewer than 2,000 cubic meters per year shall be shipped out of the State of Idaho.

2. The sole remedy for failure by DOE to meet any of these deadlines or requirements shall be the suspension of DOE spent fuel shipments to INEL as set forth in Section K.1.

C. SPENT FUEL & HIGH-LEVEL WASTE SHIPMENTS LEAVING IDAHO

1. DOE shall remove all spent fuel, including naval spent fuel and Three Mile Island spent fuel from Idaho by January 1, 2035. Spent fuel being maintained for purposes of testing shall be excepted from removal, subject to the limitations of Section F.1 of this Agreement.

2. Until all of the aluminum-clad spent fuel then stored at INEL has been shipped to the Savannah

River Site, the cumulative number of shipments of spent fuel from the Savannah River Site to INEL under Section D as of the end of any calendar year shall not exceed the cumulative number of shipments of aluminum-clad spent fuel from INEL to the Savannah River Site for the same period.

3. DOE shall treat all high-level waste currently at INEL so that it is ready to be moved out of Idaho for disposal by a target date of 2035.

D. SHIPMENTS OF SPENT FUEL TO INEL

The federal parties may transport shipments of spent fuel to INEL only in accordance with the following terms and conditions.

1. Shipments of naval spent fuel to INEL shall take place as follows:

a. The Navy may make only those shipments of naval spent fuel to INEL that are necessary to meet national security requirements to defuel or refuel nuclear powered submarines, surface warships, or naval prototype or training reactors, or to ensure examination of naval spent fuel from these sources. The Secretary of Defense, upon notice to the Governor of the State of Idaho, shall certify the total number of such shipments of naval spent fuel required to be made through the year 2035.

b. The Navy shall not ship more than twenty four (24) shipments to INEL from the date of this Agreement through the end of 1995, no more than thirty six (36) shipments in 1996, and no more than twenty (20) shipments per year in calendar years 1997 through 2000. From calendar year 2001 through 2035, the Navy may ship a running average of no more than twenty (20) shipments per year to INEL. The total number of shipments of naval spent fuel to INEL through 2035 shall not exceed 575. Shipments of naval spent fuel to INEL through 2035 shall not exceed 55 metric tons of spent fuel.

c. Prior to January 1 of each calendar year through the year 2035, the Navy shall provide to Idaho an estimate of the number of shipments and the number of metric tons of naval spent fuel to be shipped during the following calendar year.

d. By January 31 of each calendar year, the Navy shall provide to Idaho the actual number of shipments and actual number of metric tons of naval spent fuel shipped during the preceding calendar year.

e. The naval spent fuel stored at INEL on the date of the opening of a permanent repository or interim storage facility shall be among the early shipments of spent fuel to the first permanent repository or interim storage facility.

f. The sole remedy for the Navy's failure to meet any of the deadlines or requirements set forth in this section shall be suspension of naval spent fuel shipments to INEL as set forth in Section K.1.

2. Shipments of DOE spent fuel to INEL shall take place-as follows:

a. If DOE and the U.S. Department of State adopt a policy to accept spent fuel from foreign research reactors into the United States, DOE may send to INEL a maximum of 61 shipments of spent fuel from foreign research reactors during the period beginning on the date such a policy is adopted and ending on December 31, 2000. The Secretary of Energy, upon notice to the Governor of the State of Idaho, must certify that these shipments are necessary to meet national security and nonproliferation requirements. Upon such certification, DOE may ship not more than 10 such shipments from the date such policy is adopted through December 31, 1996, not more than 20 such shipments from the date the policy is adopted through December 31, 1997, and not more than 40 such shipments from the date the policy is adopted through December 31, 1998.

b. Until such time as a permanent repository or interim storage facility for storage or disposal of spent fuel, located outside of Idaho, is operating and accepting shipments of spent fuel from INEL, DOE shall be limited to shipments of spent fuel to INEL as set forth in Sections D.2.a., c., d., e., and f. After a permanent repository or interim storage facility is operating and accepting shipments of spent fuel from INEL, the State of Idaho and DOE may negotiate and reach agreement concerning the timing and number of shipments of DOE spent fuel that may be sent to INEL, in addition to those otherwise permitted under this Section D.2., for preparation for storage or disposal outside the State of Idaho.

c. After December 31, 2000, DOE may transport shipments of spent fuel to INEL constituting a total of no more than 55 metric tons of DOE spent fuel (equivalent to approximately 497 truck shipments) and subject to the limitations set forth in Sections D.2.e., f., g., and h. below, except that the limitations of Section-D:2.a. above will not apply.

d. No shipments of spent fuel shall be made to INEL from Fort St. Vrain, unless a permanent repository or interim storage facility for spent fuel located outside of Idaho has opened and is accepting spent fuel from INEL, in which case such shipments may be made for the purpose of treating spent fuel to make it suitable for disposal or storage in such a repository or facility. Shipments of spent fuel from Fort St. Vrain shall remain at INEL only for a period of time sufficient to allow treatment for disposal or storage in such a repository or facility. The total number of Fort St. Vrain shipments shall not exceed 244, constituting no more than sixteen (16) metric tons of spent fuel, and shall be in addition to those allowed under Section D 2.c. above.

e. Except as set forth in Section D.2.d. above, DOE will make no shipments of spent fuel from commercial nuclear power plants to INEL.

f. After December 31, 2000, and until an interim storage facility or permanent repository is opened and accepting spent fuel from INERT, DOE shall not ship to INEL more than 20 truck shipments of spent fuel in any calendar year, except that:

(i) In one calendar year only, DOE may make not more than 83 truck shipments of spent fuel to INEL from the West Valley Demonstration Project;

(ii) DOE may not make more than 13 truck shipments in any of the nine calendar years succeeding the shipment of the West Valley Demonstration Project spent fuel to INEL; and

(iii) Shipments DOE is entitled to make to INEL in any calendar year, but has not made, may be shipped in any subsequent calendar year, notwithstanding the limitations in this Section D.2.f. on the number of shipments per year.

For purposes of this section and Section D.2.c., in determining the number of truck shipments, one rail shipment shall be deemed equivalent to 10 truck shipments, except that in the case of shipments from West Valley Demonstration Project, seven rail shipments shall be deemed to be equal to 83 truck shipments. DOE may elect to make rail shipments in lieu of truck shipments, in accordance with this conversion formula and subject to other limitations of this section.

g. Prior to January 1 of each calendar year through the year 2035, DOE shall provide to Idaho an estimate of the number of shipments and the number of metric tons of DOE spent fuel to be shipped during the following calendar year.

h. No later than January 31st of each calendar year, DOE shall provide to Idaho the actual number of shipments and actual number of metric tons of DOE spent fuel shipped during the preceding year.

i. The sole remedy for DOE's failure to meet any of the deadlines or requirements set forth in this section shall be the suspension of DOE spent fuel shipments to INEL as set-forth in Section K.1.

E. TREATMENT & TRANSFER OF EXISTING WASTES AT INEL

1. Treatment Commitment. DOE agrees to treat spent fuel, high-level waste, and transuranic wastes in Idaho requiring treatment so as to permit ultimate disposal outside the State of Idaho.

2. Mixed Waste Treatment Facility. DOE shall, as soon as practicable, commence the procurement of a treatment facility ("Facility") at INEL for the treatment of mixed waste, transuranic waste and alpha-emitting mixed low-level waste ("Treatable Waste"). DOE shall execute a procurement contract for the Facility by June 1, 1997, complete construction of the Facility by December 31, 2002, and commence operation of the Facility by March 31, 2003. Commencement of construction is contingent upon Idaho approving necessary permits.

a. Treatment of Non-INEL Wastes. Any and all Treatable Waste shipped into the State of Idaho for treatment at the Facility shall be treated within six months of receipt at the Facility, with the exception of two cubic meters of low-level mixed waste from the Mare Island Naval Shipyard which will complete base closure for nuclear work in 1996. DOE may request an exception to the six month time period on a case-by-case basis, considering factors at the shipping site such as health and safety concerns, insufficient permitted storage capacity, and base or site closures. Any transuranic waste received from another site for treatment at the INEL shall be shipped outside of Idaho for storage or disposal within six months following treatment. DOE shall continue to use the Federal Facility Compliance Act process, as facilitated by the National Governors' Association, to determine what locations are suitable for mixed low-level waste treatment and storage.

3. Operation of High-Level waste Evaporator. DOE shall commence operation of the high-level waste evaporator by October 31, 1996; and operate the evaporator in such a manner as to reduce the tank farm liquid waste volume by no fewer than 330,000 gallons by December 31, 1997. Efforts will continue to reduce the remaining volume of the tank farm liquid waste by operation of the high-level waste evaporator.

4. Calcination of Remaining Non-Sodium Bearing Liquid Wastes. DOE shall complete the process of calcining all the remaining non-sodium bearing liquid high-level wastes currently located at INEL by June 30, 1998.

5. Calcination of Sodium-Bearing Wastes. DOE shall commence calcination of sodium-bearing liquid high-level wastes by June 1, 2001. DOE shall complete calcination of sodium-bearing liquid high-level wastes by December 31, 2012.

6. Treatment of Calcined Wastes. DOE shall accelerate efforts to evaluate alternatives for the treatment of calcined waste so as to put it into a form suitable for transport to a permanent repository or interim storage facility outside Idaho. To support this effort, DOE shall solicit proposals for feasibility studies by July 1, 1997. By December 31, 1999, DOE shall commence negotiating a plan and schedule with the State of Idaho for calcined waste treatment. The plan and schedule shall provide for completion of the treatment of all calcined waste located at INEL by a date established by the Record of Decision for the Environmental Impact Statement that analyzes the alternatives for treatment of such waste. Such Record of Decision shall be issued not later than December 31, 2009. It is presently contemplated by DOE that the plan and schedule shall provide for the completion of the treatment of all calcined waste located at INEL by a target date of December 31, 2035. The State expressly reserves its right to seek appropriate relief from the Court in the event that the date established in the Record of Decision for the Environmental Impact Statement that analyzes the alternatives for treatment of such waste is significantly later than DOE's target date. In support of the effort to treat such waste, DOE shall submit to the State of Idaho its application for a RCRA (or statutory equivalent) Part B permit by December 1, 2012.

7. Transfer of Three Mile Island Fuel. DOE shall complete construction of the Three Mile Island dry

storage facility by December 31, 1998. DOE shall commence moving fuel into the facility by March 31, 1999, and shall complete moving fuel into the facility by June 1, 2001.

8. Transfer out of Wet Storage. By December 31, 1999, DOE shall commence negotiating a schedule with the State of Idaho for the transfer of all spent fuel at INEL out of wet storage facilities. DOE shall complete the transfer of all spent fuel from wet storage facilities at INEL by December 31, 2023. If DOE determines that transfer to dry storage of any portion of such spent fuel is technically infeasible, or that transfer to such dry storage presents significantly greater safety or environmental risks than keeping the fuel in wet storage, DOE shall inform the State and propose a later date or alternative action. If the State does not agree to such later date or alternative action, DOE may apply to the Court for appropriate relief. DOE shall, after consultation with the State of Idaho, determine the location of the-dry storage facilities within INEL, which shall, to the extent technically feasible, be at a point removed from above the Snake River Plain Aquifer ("Aquifer").

9. The sole remedy for DOE's failure to meet any of the deadlines or requirements set forth in this section shall be the suspension of DOE spent fuel shipment to INEL as set forth in Section K.1.

F. SPENT FUEL PROGRAM

1. Establishment of INEL as DOE Spent Fuel Lead Laboratory. DOE shall, within thirty days of entry of this Agreement as a court order, designate INEL as the Department's lead laboratory for spent fuel. DOE shall direct the research, development and testing of treatment, shipment and disposal technologies for all DOE spent fuel, and all such DOE activities shall be coordinated and integrated under the direction of the Manager, DOE-Idaho Operations Office. Such designation shall not permit the shipment to INEL of any spent fuel beyond that permitted by this Agreement with the exception that quantities of spent fuel brought to INEL for testing in excess of those permitted by this Agreement shall leave the State of Idaho within five years of the date of receipt at INEL.

2. Construction of Dry Storage. DOE shall include in its appropriation request for federal fiscal year 1998 to the Executive Office of the President funds necessary for DOE to initiate the procurement of dry storage at INEL to replace wet, below ground facilities. Spent fuel loading into dry storage shall commence by July 1, 2003.

3. Funding for Dry Cell Expansion Project. The Naval Nuclear Propulsion Program shall include in its appropriation request to the Executive Office of the President for federal fiscal year 1997 funds necessary for the Dry Cell Expansion Project ("Project") at the Expended Core Facility at the Naval Reactors Facility to accommodate removal of excess material and examination of naval spent fuel in a dry condition. The Project shall commence as soon as Idaho issues the required permit under the Clean Air Act and funding is appropriated. Completion of this project shall result in the expenditure of approximately $26 million dollars over the next five years.

4. Multi-Purpose Canisters. DOE and the Navy shall employ Multi-Purpose Canisters ("MPCs") or comparable systems to prepare spent fuel located at INEL for shipment and ultimate disposal of such fuel outside Idaho. Procurement shall be performed in accordance with the Federal Acquisition Regulation which ensures that companies in Idaho will have opportunity to bid on and obtain any competitive contracts for such work. The Record of Decision on the NEPA analysis shall be completed by April 30, 1999.

5. ECF Hot Cell Facility Upgrade. The Naval Nuclear Propulsion Program shall include in its appropriation request for federal fiscal year 1997 to the Executive Office of the President funds necessary to proceed with upgrades which shall require approximately $12 million of expenditures during the next three years.

6. ECF Dry Storage Container Loading Station. The Naval Nuclear Propulsion Program shall include

in its appropriation request for federal fiscal year 1997 to the Executive Office of the President funds necessary to proceed with design and construction of a dry storage container loading station at ECF. This project shall require no less than $20 million of expenditures during the next five years.

7. Funding for Discretionary Environmental Remediation Work at the Naval Reactors Facility. The Naval Nuclear Propulsion Program shall undertake environmental remediation efforts at the Naval Reactors Facility totaling approximately $45 million over the next five years.

8. Water Pool Reracking. DOE may proceed with installing new racks into the water pool in the building at the Idaho Chemical Processing Plant Facility currently holding naval spent fuel to provide enhanced capability for spent fuel storage in the existing water pool space until dry storage can be made available. Installation of the new racks may commence as soon as Idaho issues the necessary permit under the Clean Air Act. Idaho shall issue said permit within 180 days after DOE re-submits its application to Idaho.

G. INEL ENVIRONMENTAL RESTORATION PROGRAM

1. INEL Environmental Restoration Program to Continue. DOE shall continue to implement the INEL environmental restoration program in coordination with Idaho and EPA. Such implementation shall be consistent with the schedules contained in the Federal Facilities Agreement and Consent Order (FFA/CO) entered into with the State of Idaho, EPA and DOE, and it shall include schedule requirements developed pursuant to the completed and future Records of Decision under the FFA/CO. The sole remedies for failure to implement the environmental restoration activities specified in the FFA/CO shall be those specified in the FFA/CO.

H. OBTAINING TIMELY FEDERAL FUNDING FOR COMPLIANCE WITH THIS ORDER

1. Compliance Funding. DOE and the Naval Nuclear Propulsion Program shall share budget information concerning INEL with Idaho prior to submitting the budget request to the Executive Office of the President. Consultations with the State of Idaho shall continue throughout the budget process. The current DOE estimate for the costs of the activities and projects described in Sections A through G over the next five years is approximately $200 million above established budget targets.

I. FEDERAL FINDS FOR THIS SETTLEMENT AGREEMENT

1. DOE shall provide to the State of Idaho beginning in federal fiscal year 1996 and continuing through 1997-2000, a total amount of $30 million for community transition purposes and any other purposes that are mutually acceptable to the parties, such as the non-Federal development of Boron Neutron Capture Therapy and Radiological Toxicology technology in Idaho.

2. Acoustic Research Funding. The Navy shall include in its appropriation request to the Executive Office of the President for federal fiscal year 1997 no less than $7 million for the Navy to construct a Ships Model Engineering and Support Facility at the Naval Surface Warfare Center, Carderock Division, Acoustic Research Detachment at Bayview, Idaho.

J. GOOD FAITH COMPLIANCE & AFFIRMATIVE SUPPORT

1. The federal parties and Idaho agree that the activities to be performed under this Agreement and the subsequent Consent Order are in the public interest. The federal parties and Idaho acknowledge the complexity of this Agreement and have agreed to act in good faith to effectuate its fulfillment. The federal parties and Idaho shall affirmatively support this Agreement and its terms, conditions, rights and obligations in any administrative or judicial proceeding. The federal parties and Idaho intend to seek a sense of the Congress resolution expressing support for the terms, conditions, rights and obligations contained in this Agreement- and the subsequent Consent Order and recommending to

future Congresses that funds requested by the President to carry out this Agreement be appropriated. In any administrative or judicial proceeding, Idaho shall support the adequacy of the EIS and ROD against any challenges by third parties. Idaho shall have the ability, in its sole discretion, to waive performance by the federal parties of any terms, conditions and obligations contained in this Agreement.

2. Idaho shall promptly issue, upon submission of legally sufficient applications, all permits, licenses or other approvals needed by the DOE, the Navy or the Naval Nuclear Propulsion Program for the performance of any of their respective obligations set forth in this Agreement.

3. No provision of this Agreement shall compel any party to act without due legal authority. Performance by every party under this Agreement shall be subject to and comply with all applicable federal statutes, regulations and orders, including the Anti-Deficiency Act. The inability of any party to comply with the provisions of this Agreement, or a delay in such compliance, as a result of any applicable federal statute, regulation or order shall not subject that party to judicial enforcement under Section K.2.a, but shall not preclude the application of Sections K.1.a. or K.1.b.

4. In the event any required NEPA analysis results in the selection after October 16, 1995, of an action which conflicts with any action identified in this Agreement, DOE or the Navy may request a modification of this Agreement to conform the action in the Agreement to that selected action. Approval of such modification shall not be unreasonably withheld. If the State refuses to accept the requested modification, DOE or the Navy may seek relief from the Court. On motion of any party, the Court may extend the time for W E or the Navy to perform until the Court has decided whether to grant relief. If the Court determines that the State has unreasonably withheld approval, the Agreement shall be conformed to the selected action. If the Court determines that the State has reasonably withheld approval, the time for DOE or the Navy to perform the action at issue shall be as set forth in this Agreement and subject to enforcement as set forth section in Section K.1.

5. Effect of Certain Court Orders.

a. Navy. In the event that a court order is entered in the case of Snake River Alliance Education Fund v. United States Department of Energy, No. CV-95-0331-S-EJL (D. Idaho), or in any other judicial proceeding, that prohibits in whole or in part any shipment of spent fuel to INEL by the Navy under section D, then all obligations, requirements and deadlines of the federal parties under this Agreement shall be suspended during the period of applicability of the order. Upon the vacating, dissolving or reversing of any such order, the obligations, deadlines and requirements provided for in this Agreement shall be extended by a period that corresponds to their period of suspension.

b. DOE. In the event that a court order is entered in the case of Snake River Alliance Education Fund v. United States Department of Energy, No. CV-95-0331-S-EJL (D. Idaho), or in any other judicial proceeding, that prohibits in whole or in part any shipment of spent fuel to INEL by DOE under section D, then the DOE has the option to suspend all DOE shipments to INEL and suspend all of DOE's obligations, requirements and deadlines under this Agreement during the period of applicability of the order. If DOE exercises this option, then upon the vacating, dissolving, or reversing of any such order, DOE's obligations, deadlines and requirements provided for in this Agreement shall be extended by a period that corresponds to their period of suspension.

K. ENFORCEMENT

1. Succession of Shipments.

a. DOE. If DOE fails to satisfy the substantive obligations or requirements it has agreed to in this Agreement or fails to meet deadlines for satisfying such substantive obligations or requirements, shipments of DOE spent fuel to INEL shall be suspended unless and until the parties agree or the

Court determines that such substantive obligations or requirements have been satisfied. .

b. Navy. If the Navy or the Naval Nuclear Propulsion Program fails to satisfy the substantive obligations or requirements it has agreed to in this Agreement or fails to meet deadlines for satisfying such substantive obligations or requirements, shipments of Navy spent fuel to INEL shall be suspended unless and until the parties agree or the Court determines that such substantive obligations or requirements have been satisfied.

2. Other Enforcement

a. Judicial Enforcement. The Court may enforce the rights, obligations and requirements assigned by this Agreement, other than those exclusively enforceable under Section K.1., pursuant to all legal and equitable remedies available to the courts of the United States, including, but not limited to, use of the Court's contempt powers.

b. RCRA Enforcement. Nothing in this Agreement shall prohibit the State of Idaho from requiring necessary remedial actions as set forth in the Resource Conservation and Recovery Act, 42 U.S.C. section 6929 ("RCRA") (or statutory equivalent), including penalty and fine procedures, the sums of which shall be payable to the State of Idaho.

c. Payment Obligation. In the event that the federal parties do not carry out the requirement that all spent fuel located at INEL be removed from Idaho by January 1, 2035, then subject to the availability of the appropriations provided in advance for this purpose, the federal parties shall pay to the State of Idaho $60,000 for each day such requirement has not been met.

3. Prior Orders, Agreements end Decisions. The terms of this Agreement shall supersede all rights, duties and obligations set forth in any prior orders, agreements or decisions entered in this litigation, captioned Public Service Company of Colorado v. Batt, and United States of America v. Batt, Nos. CV 91-0035-S-EJL and CV 91-0054-S-EJL, except for the provisions of paragraph 4 of the December 22, 1993 Court Order.

4. Dispute Resolution. In the event that any party to this Agreement contends that any other party has violated any terms of the Agreement, the parties shall seek to resolve their differences informally before asking for resolution by the Court.

L. CONSENT ORDER

1. The parties agree they shall jointly present this Agreement to the U.S. District Court with a proposed Consent Order which will provide for the incorporation of this Agreement, continuing jurisdiction of the Court and the administrative termination of this action without prejudice to the right of the parties to reopen the proceedings for good cause shown. This Agreement and Consent Order shall not preclude any party from applying to the Court under Rule 60, of the Federal Rules of Civil Procedure, or the Court from granting relief thereunder.

2. If the Consent Order is not entered by the Court, in accordance with Section L.1 above, within 45 days of lodging with the Court, then either party to this Agreement may elect to terminate this Agreement, in which case this Agreement becomes null and void, and of no force or effect.

For the Federal Parties:

/s/
Robert R Nordhaus
General Counsel
Department of Energy

/s/
Thomas P. Grumbly
Assistant Secretary for Environmental Management
Department of Energy

/s/
Steven S Honigman
General Counsel
Department of the Navy

/s/
Admiral Bruce DeMars
Director, Naval Nuclear Propulsion Program

For the State of Idaho

/s/
Philip E Batt
Governor, State
State of Idaho

/s/
Alan G Lance
Attorney General,
State of Idaho

Appendix L

Contractor Disclosure Statement

NEPA DISCLOSURE STATEMENT FOR PREPARATION OF EIS FOR THE TREATMENT AND MANAGEMENT OF SODIUM-BONDED SPENT NUCLEAR FUEL

CEQ regulations at 40 CFR 1506.5(c), which have been adopted by DOE (10 CFR 1021), require contractors who will prepare an EIS to execute a disclosure specifying that they have no financial or other interest in the outcome of the project. The term "financial interest or other interest in the outcome of the project," for the purposes of this disclosure, is defined in the March 23, 1981 guidance "Forty Most Asked Questions Concerning CEQ's National Environmental Policy Act Regulations," 46 FR 18026-18038 at Question 17a and b.

"Financial or other interest in the outcome of the project 'includes' any financial benefit such as a promise of future construction or design work in the project, as well as indirect benefits the contractor is aware of (e.g., if the project would aid proposals sponsored by the firm's other clients)." 46 FR 18026-18038 at 18031.

In accordance with these requirements, the offeror and any proposed subcontractors hereby certify as follows: (check either (a) or (b) to assure consideration of your proposal)

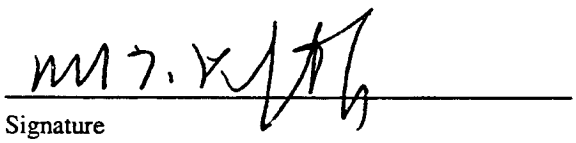
(a) ___✓___ Offeror and any proposed subcontractor have no financial interest in the outcome of the project.

(b) _____ Offeror and any proposed subcontractor have the following financial or other interest in the outcome of the project and hereby agree to divest themselves of such interest prior to award of this contract.

Financial or Other Interests:

1.
2.
3.

Certified by:



Signature

Richard T. Profant
Name

Corporate Vice President
Integrated Environmental Services Operation

July 14, 2000
Date

Science Applications International Corporation