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FINAL

ENVIRONMENTAL IMPACT STATEMENT

Management of Commercially Generated Radioactive Waste

Volume 1

October 1980

U.S. Department of Energy
Assistant Secretary for Nuclear Energy
Office of Nuclear Waste Management
Washington, D.C. 20545

FOREWORD

In his February 12, 1980, message to Congress, the President of the United States announced a comprehensive program for management of radioactive waste. With regard to waste disposal, the President said:

> ". . . for disposal of high-level radioactive waste, I am adopting an interim planning strategy focused on the use of mined geologic repositories capable of accepting both waste from reprocessing and unreprocessed commercial spent fuel. An interim strategy is needed since final decisions on many steps which need to be taken should be preceded by a full environmental review under the National Environmental Policy Act. In its search for suitable sites for high-level waste repositories, the Department of Energy has mounted an expanded and diversified program of geologic investigations that recognizes the importance of the interaction among geologic setting, repository host rock, waste form, and other engineered barriers on a site-specific basis. Immediate attention will focus on research and development and on locating and characterizing a number of potential repository sites in a variety of different geologic environments with diverse rock types. When four to five sites have been evaluated and found potentially suitable, one or more will be selected for development as a licensed, full-scale repository."

In an accompanying Fact Sheet issued by the White House Press Secretary it was noted that the President will reexamine this interim strategy and decide whether any changes need to be made following completion of the necessary environmental reviews as required by the National Environmental Policy Act (NEPA). Issuance of this environmental impact statement (EIS) is intended to serve as a basis for that reexamination.

In keeping with the mandate of NEPA, this EIS analyzes the significant environmental impacts that could occur if various technologies for management and disposal of high-level and transuranic wastes from commercial nuclear power reactors were to be developed and implemented. This EIS will serve as the environmental input for the decision on which technology, or technologies, will be emphasized in further research and development activities in the commercial waste management program.

The action proposed in this EIS is to 1) adopt a national strategy to develop mined geologic repositories for disposal of commercially generated high-level and transuranic radioactive waste (while continuing to examine subseabed and very deep hole disposal as potential backup technologies) and 2) conduct an R&D program to develop such facilities and the necessary technology to ensure the safe long-term containment and isolation of these wastes.

The Department has considered in this Statement:

- Development of conventionally mined deep geologic repositories for disposal of spent fuel from nuclear power reactors and/or radioactive fuel reprocessing wastes.[(a)]
- Balanced development of several alternative disposal methods.
- No waste disposal action.

Prior to announcing his national waste management program, the President received recommendations on the program from an Interagency Review Group whose report was issued in April 1979. In their report, the Interagency Review Group analyzed a number of possible strategies for the program of high-level waste disposal. These strategies differed with regard to the number of diverse sites that should be examined in a geologic disposal program prior to construction of a facility and in one case discussed the implementation of technologies other than mined geologic repositories.

This EIS has not specifically examined the strategies reviewed by the Interagency Review Group but the essential differences between them are covered in the comparison of the first two program alternatives considered here. These alternatives have been examined for a number of different scenarios of future nuclear power use and for a range of times for operation of facilities, including those considered by the Interagency Review Group.

A draft of this environmental impact statement--"Management of Commercially Generated Radioactive Waste"--was issued for review and comment as DOE/EIS-0046D on April 20, 1979. Copies were sent to Federal agencies with responsibilities associated with radioactive waste disposal, to governors of all states, and to public interest groups known to have an interest in waste management. Comments were received from the following Federal agencies:

Department of Commerce
Department of Health, Education and Welfare
Department of the Interior
Environmental Protection Agency
Federal Energy Regulatory Commission
Nuclear Regulatory Commission
and from agencies or officials from 17 states.

A total of 219 written communications, incorporating about 2000 comments, were received and considered in preparation of this final Statement.

An impartial Hearing Board, composed of specialists in several fields, was appointed to conduct a series of public hearings on the draft Statement. The board members had not been DOE personnel nor previously involved with the DOE waste management program and were employed specifically to conduct the hearings and evaluate the public concerns. Hearings were held in Washington, D.C.; Chicago, Illinois; Atlanta, Georgia; Dallas, Texas; and San

(a) The Statement does not formally consider radioactive wastes related to defense programs; however, in a generic sense, systems that can safely dispose of commercial radioactive wastes are expected to safely dispose of defense wastes.

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CHAPTER 1

SUMMARY

In the course of producing electrical power in light water reactors (LWRs), the uranium fuel accumulates fission products until the fission process is no longer efficient for power production. At that point the fuel is removed from the reactor and stored in water basins to allow radioactivity to partially decay before further disposition. This fuel is referred to as "spent fuel." Although spent fuel as it is discharged from a reactor is intensely radioactive, it has been stored safely in moderate quantities for decades. Spent fuel could be reprocessed, and about 99.5% of the remaining uranium and newly formed plutonium could be recovered for reuse. However, present policy dictates that spent LWR fuel reprocessing will be indefinitely deferred because of concern that widespread separation of plutonium could lead to proliferation of nuclear weapons. As a result, spent fuel is currently stored for possible future reprocessing or disposal. Storage or disposal must be designed so that nuclear waste will not be a present or future threat to public health and safety.

The United States Department of Energy (DOE) has the responsibility to develop technologies for management and disposal of certain classes of commercially generated radioactive wastes (namely high-level and transuranic).[a] High-level waste is defined as either the aqueous solution from the first-cycle solvent extraction, where spent fuel is reprocessed for recycle of uranium and plutonium, or spent fuel if disposed of. High-level waste is also intensely radioactive.

Other wastes are generated during reprocessing that, although larger in volume than high-level wastes, are less intensely radioactive. Wastes that contain more than a specified amount of radionuclides of atomic number greater than that of uranium are called transuranic (TRU) wastes. TRU wastes are categorized here as either remotely handled (RH) or contact-handled (CH) wastes, depending on the requirements for radiation protection of personnel. Special attention must be given to TRU wastes because they contain alpha particle-emitting nuclides that are of particular concern as a result of their long half lives and tenacious retention if incorporated in the body. Other waste forms that include neither high-level nor TRU are so-called low-level wastes.[b]

The principal objective of waste disposal is to provide reasonable assurance that these wastes, in biologically significant concentrations, will be permanently isolated from the human environment. To provide input to the decision on a planning strategy for disposal of these radioactive wastes, this Statement presents an analysis of environmental impacts that could occur if various technologies for management and disposal of such wastes were to be developed and implemented.

(a) In a message to Congress on February 12, 1980, the President reiterated the role of DOE as lead agency for management and disposal of radioactive wastes.
(b) Low level wastes, other than those originating at DOE facilities, are managed and disposed of by licenses in accordance with regulations of the NRC.

The DOE is proposing a program strategy emphasizing development of conventionally mined waste repositories, deep in the earth's geologic formations, as a means of disposing of commercially-generated high-level and TRU wastes. Adoption of this program strategy constitutes a major federal action for which the National Environmental Policy Act of 1969 (NEPA) requires preparation of a detailed environmental impact statement (EIS).

This summary highlights the major findings and conclusions of this final Statement. It reflects the public review of and comments offered on the draft Statement. Included are descriptions of the characteristics of nuclear waste, the alternative disposal methods under consideration, and potential environmental impacts and costs of implementing these methods. Because of the programmatic nature of this document and the preliminary nature of certain design elements assumed in assessing the environmental consequences of the various alternatives, this study has been based on generic, rather than specific, systems. At such time as specific facilities are identified for particular sites, statements addressing site-specific aspects will be prepared for public review and comment.

1.1 THE NEED FOR WASTE MANAGEMENT AND DISPOSAL

There are now about 70 operating commercial LWR power reactors in the United States, which represent approximately 50 GWe[(a)] of installed nuclear powered electrical generating capacity. The amounts of spent fuel accumulated for the present (1980) inventory and for alternative nuclear power generating scenarios considered in this Statement are shown in Table 1.1.1.

TABLE 1.1.1. Total Spent Fuel Disposal or Reprocessing Requirements

Nuclear Power Growth Assumption			
Case	Scenario	Energy Generated, GWe-yr[(a)]	Spent Fuel Discharged, MTHM[(b)]
1	Present Inventory Only--Reactors Shut Down in 1980[(c)]	200	10,000
2	Present Capacity (50 GWe)[(c)] and Normal Reactor Life	1,300	48,000
3	250 GWe System by Year 2000 and Normal Reactor Life (No new reactors after Year 2000)[(d)]	6,400	239,000
4	250 GWe System by Year 2000 and Steady State Capacity to Year 2040 (New reactors to maintain output)[(d)]	8,700	316,000
5	500 GWe System by Year 2040[(d)]	12,100	427,000

(a) Energy generated is based on the total accumulated through the year 2040.
(b) MTHM = metric tons (1000 kg = about 1.1 U.S. tons) of heavy metal in original fuel. One MTHM of spent fuel consists of about 96% uranium, 1% plutonium and 3% fission products.
(c) Reprocessing is not applicable to Cases 1 and 2 because in Case 1 there is no need for reprocessing and in Case 2 no economic incentives exist for reprocessing.
(d) Waste management impacts of nuclear power generation through the year 2040 are considered for these scenarios.

The total radioactivity in one MTHM of LWR fuel and equivalent HLW for various times after discharge from a reactor is shown in Figure 1.1.1. Similarly, the heat generation rate in this fuel is illustrated in Figure 1.1.2. These figures show that a reduction by a factor of about 1,000 in radioactivity relative to one-year-old fuel is reached in about 700 years for spent fuel and in about 200 years for uranium and plutonium recycle high-level waste. The heat generation rate is lower by a factor of 100 for spent fuel at about 300 years and for recycle high-level waste at about 150 years.

(a) One GWe = 1×10^9 watts.

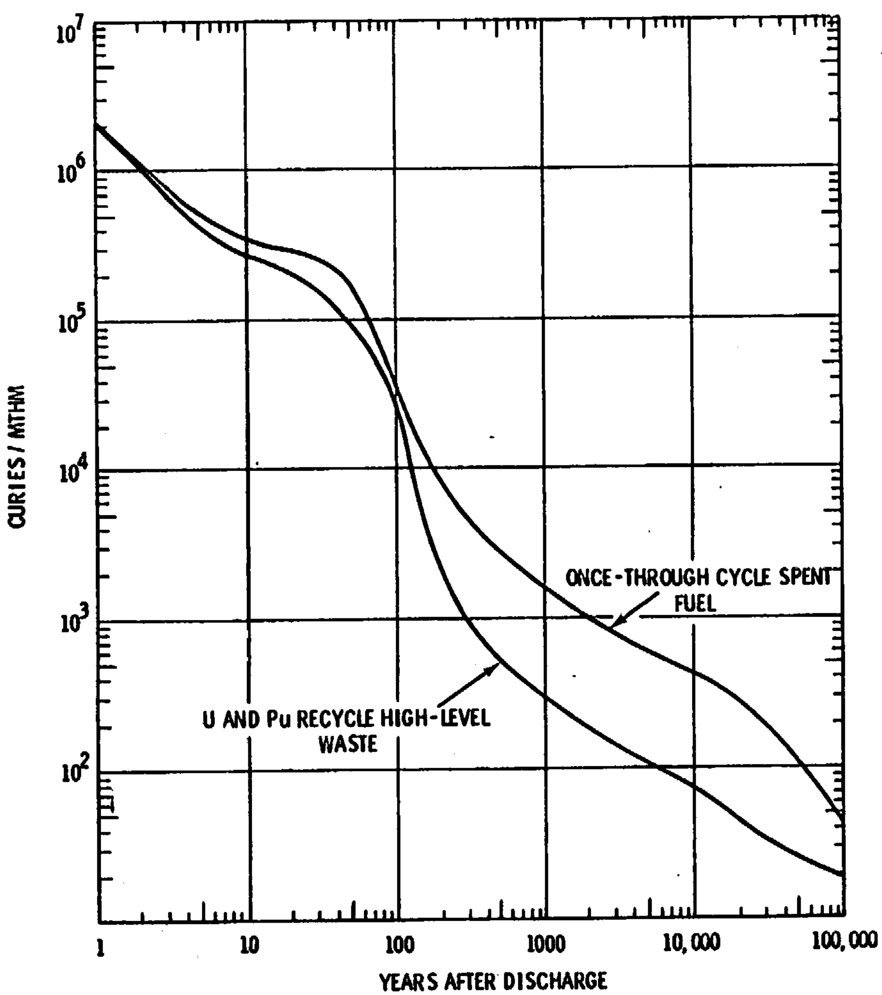

FIGURE 1.1.1. Radioactivity in Spent Fuel and High-Level Waste as a Function of Time

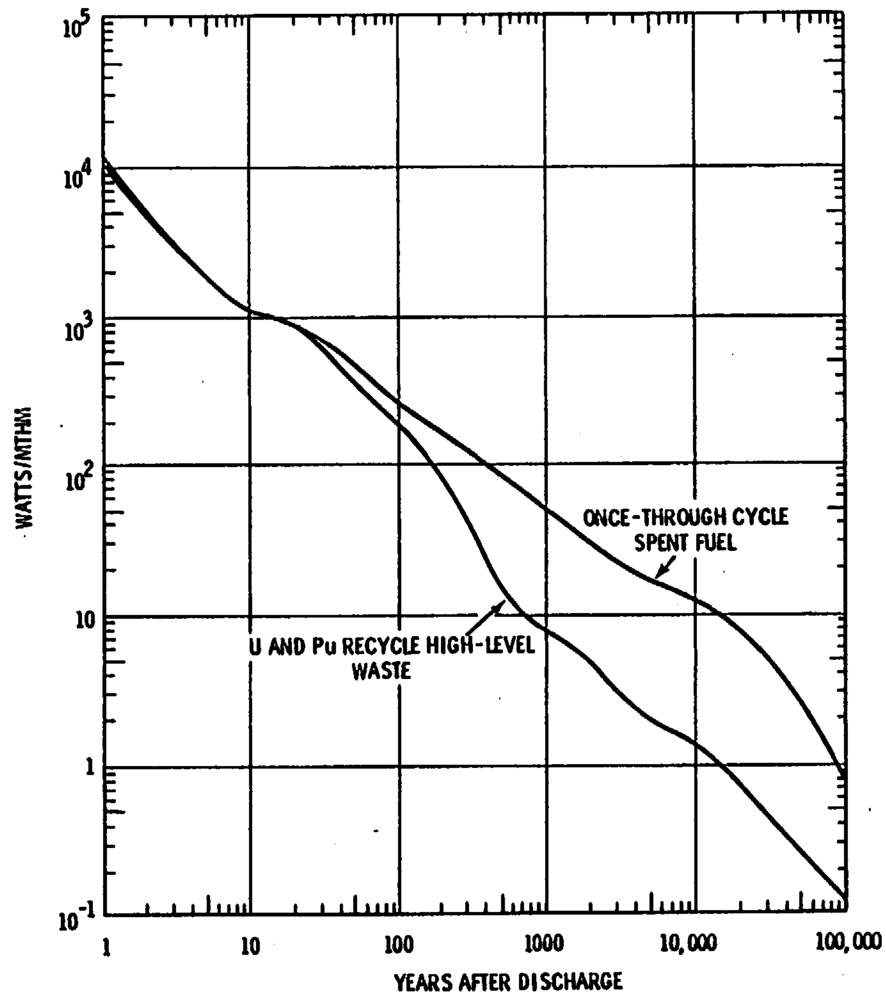

FIGURE 1.1.2. Heat Generation Rate of Spent Fuel and High-Level Waste as a Function of Time

The President, in his February 12, 1980 message on radioactive wastes, called for waste disposal facilities that could receive wastes from both the commercial nuclear power production program and the national defense program. Since defense wastes are not explicitly treated in this Statement, it is not intended to provide environmental input for disposal decisions on defense wastes. However, in a generic sense, systems that can adequately dispose of commercial radioactive wastes can reasonably be expected to adequately dispose of defense wastes, since the processed wastes from the national defense program produce lower temperatures and lower radiation intensities than do wastes from the same quantity of similarly processed commercial fuel. Thus, assuming that other factors are equal, repository loading criteria would generally be less stringent (in terms of quantities of waste per unit area) for defense wastes than for commercial wastes. For this reason certain of the analyses of impacts presented in this EIS should be of use in the preparation of EIS's on the long term management of high-level and TRU defense waste.

1.2 THE PROGRAMMATIC ALTERNATIVES

The programmatic alternatives considered in this Statement are:

- Proposed Action. The research and development program for waste management will emphasize use of mined repositories in geologic formations in the continental U.S. capable of accepting radioactive wastes from either the once-through or reprocessing cycles (while continuing to examine subseabed and very deep hole disposal as potential backup technologies). This action will be carried forward to identify specific locations for the construction of mined repositories. The proposed action does not preclude further study of other disposal techniques. For example, the selective use of space disposal for specific isotopes might be considered.

- Alternative Action. The research and development program would emphasize the parallel development of several disposal technologies. This action implies an R&D program to bring the knowledge regarding two or three disposal concepts and their development status to an approximately equal level. Based upon the Department of Energy's current evaluation, the likely candidate technologies for this parallel development strategy would be:
  1) geologic disposal using conventional mining techniques
  2) placement in sediment beneath the deep ocean (subseabed)
  3) disposal in very deep holes.

  At some later point, a preferred technology would be selected for construction of facilities for radioactive waste disposal.

- No Action Alternative. This alternative would eliminate or significantly reduce the Department of Energy's research and development programs for radioactive waste disposal. Under this alternative, existing spent fuel would be left indefinitely where it is currently stored and any additional spent fuel discharged from future operation of commercial nuclear power plants would likewise be stored indefinitely in water basin facilities either at the reactors or at independent sites.

1.3 THE PROPOSED ACTION

The proposed action is to select and pursue a programmatic strategy that would lead to disposal of existing and future commercially generated radioactive high-level and transuranic wastes in mined repositories in geologic formations. This Statement addresses environmental impacts related to implementing such disposal[a]. The programmatic strategy will direct effort and concentrate resources on a research and development program leading to repositories and to site-selection processes. Some support will be provided to further evaluate the alternatives of subseabed disposal and disposal in very deep holes.

Environmental impacts related to repository construction, operation, and decommissioning are analyzed in this Statement as are the impacts of predisposal waste treatment, storage and transportation to the extent they might effect selection of a disposal option. Environmental impacts are developed for individual example facilities and for systems based on the power growth scenarios described in Table 1.1.1 This very broad or generic approach to evaluating the environmental issues provides a comprehensive overview of the likely consequences of the proposed action and constitutes the first phase of DOE's NEPA implementation plan for waste management and disposal (DOE/NE-0007 1980). This plan for waste management and disposal is based on a tiered approach, which is designed to eliminate repetitive discussions on the same issues and to focus on important issues ready for decision at each level of environmental review. Thus, as more site- or facility-specific decision points are approached, and before each such decision and before conducting of activities that may cause an adverse impact or limit the choice of reasonable alternatives, additional environmental assessments, or impact statements will be prepared as appropriate.

The proposed research and development program for waste management will emphasize use of mined repositories in geologic formations capable of accepting radioactive wastes from either the once-through or reprocessing cycles. This program will be carried forward to identify specific locations for the construction of mined repositories.

Initially, site characterization programs will be conducted to identify qualified sites in a variety of potential host rock and geohydrologic settings. As qualified sites are identified by the R&D program, actions will be taken to reserve the option to use the sites, if necessary, at an appropriate time in the future. Supporting this site characterization and qualification program will be research and development efforts to produce techniques and equipment to support the placement of wastes in mined geologic repositories.

The Department of Energy proposes that the development of geologic repositories will proceed in a careful step-by-step fashion. Experience and information gained in each phase of the development program will be reviewed and evaluated to determine if there is sufficient knowledge to proceed to the next stage of development and research. The Department plans to proceed on a technically conservative basis allowing for ready retrievability of the emplaced waste for some initial period of time.

(a) Disposal of radioactive wastes in mined geologic repositories was stated by the President in his February 12, 1980 message as the interim planning strategy to receive emphasis pending environmental review under NEPA.

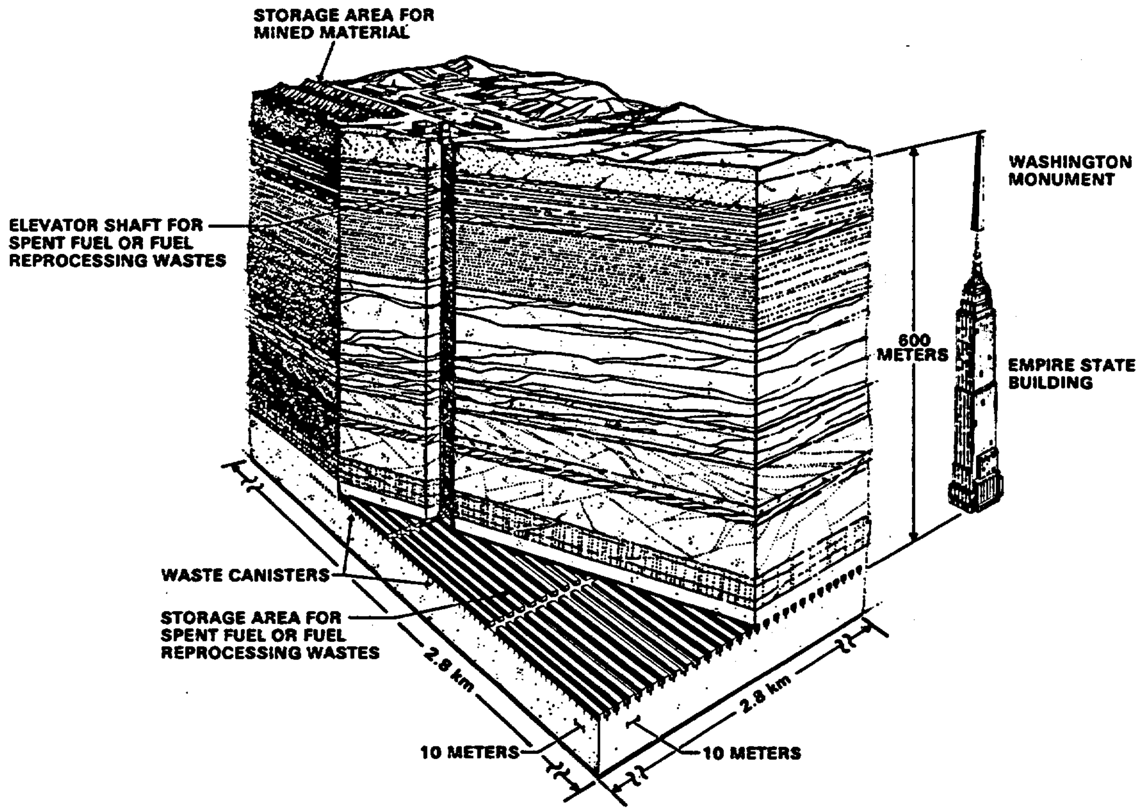

FIGURE 1.3.1 Deep Underground Geologic Waste Repository

1.3.1 Mined Geologic Disposal of Radioactive Wastes

The concept of mined geologic disposal of radioactive wastes is one in which canistered high-level wastes and other wastes in canisters, drums, boxes or other packages, as appropriate to their form, radioactive waste content and radiation intensity, are placed in engineered arrays in conventionally mined rooms in geologic formations far beneath the earth's surface. An artist's rendering of the geologic disposal concept is shown together with more familiar structures for comparison in Figure 1.3.1.

Geologic disposal, as analyzed in this Statement, also employs the concept of multiple barriers. Multiple barriers include both engineered and geologic barriers that improve confidence that radioactive wastes, in biologically significant concentrations, will not return to the biosphere. Engineered barriers include the waste form itself, canisters, fillers, overpacking, sleeves, seals and backfill materials. Each of these components may be designed to reduce the likelihood of release of radioactive material and would be selected based on site- and waste-specific considerations. Geologic barriers include the repository host rock and adjacent and overlying rock formations. While engineered barriers are tailored to a specific containment need, geologic barriers are chosen for their in-situ properties for both waste containment and isolation.

1.3.2 An Example Geologic Repository

For purposes of illustration and for estimating the environmental impacts of development and implementation of waste disposal in geologic repositories, an example repository was postulated that would have an underground area of about 800 hectares (2000 acres) and would be located about 600 meters (2000 ft) underground. This repository area provides for reasonable waste disposal capacity and is achievable from both construction and operational points of view using conventional room and pillar mining techniques. Actual repositories may be larger or smaller than 800 hectares (ha) depending upon site-specific characteristics.

In this Statement salt, granite, shale and basalt are considered as examples of repository host rock. These rock types represent a range of characteristics of candidate earth materials representative of geologic formations that might be considered but other rock types such as tuff may also be suitable candidates.

Because of restrictions of radioactive waste heat loading on the host rock (to prevent or restrict effects on the rock structure) and other structural considerations, different spacing of waste canisters (containers) would be required and would result in different repository waste capacities for a given rock type and repository area.

The number of 800-ha example repositories required for disposal of spent fuel or reprocessing wastes under the different nuclear power growth assumptions described in Section 1.1 is given in Table 1.3.1. The ranges given reflect the different load capacities (both from a permissible heat load standpoint and because of the different fractions of the 800 ha available for waste emplacement) of repositories in the different host rocks.

TABLE 1.3.1. Number of 800 Hectare Example Repositories Required for Various Nuclear Power Growth Assumptions

Case	Nuclear Power Growth Assumption	Number of Repositories: Spent Fuel	Number of Repositories: Reprocessing Wastes
1	Present Inventory Only Reactors Shut Down in 1980	0.03 to 0.1	(a)
2	Present Capacity and Normal Life	0.2 to 1	(a)
3	250 GWe System by Year 2000 and Normal Life	1 to 4	2 to 5
4	250 GWe System by Year 2000 and Steady State[(b)]	2 to 5	3 to 6
5	500 GWe System by Year 2000[(b)]	2 to 7	4 to 9

(a) If all reactors are shut down in 1980 or if nuclear power were to be restricted to present capacity there would be no economic incentive for reprocessing.
(b) Required by Year 2040.

As shown in Table 1.3.1 the subterranean area needed for spent fuel or reprocessing wastes from the power-generating scenarios considered in this Statement ranges from approximately 24 ha (60 acres) to about 7,200 ha (18,000 acres or 24 mi^2) depending upon the scenario and the choice of repository media. The larger numbers of repositories for reprocessing wastes are required principally because of the large volumes of TRU wastes requiring disposal.

Once licensing approvals are obtained, an approximate 5-year repository construction period is estimated. The operating period may range from 1 to 30 years or more depending on the size of the industry served and on the number of repositories operating concurrently.

1.3.3 Environmental Impacts Associated with Construction and Operation of Example Geologic Repositories

Environmental impacts associated with construction and operation of geologic repositories include radiological impacts, both in the short and long term, land and other resource commitments, and impacts related to ecological, nonradiological, aesthetic, and socioeconomic aspects. In the case of socioeconomic, aesthetic, and ecological impacts and hypothetical failures of repositories in the long term, impacts are summarized for a single 800-ha repository, as might be built in salt, granite, shale or basalt and containing either spent fuel or reprocessing wastes. Radiological impacts of waste management and disposal, resource commitments and dollar costs are summed in Section 1.7 for total system requirements for power growth assumptions given in Table 1.1.1.

1.3.3.1. Radiological Impacts

Radiological impacts that might be associated with repository construction (mining), operation and decommissioning, as well as those that might result from unplanned events either before or after the repository was closed were analyzed in detail. The estimated 70-year whole-body dose to a hypothetical regional population (2 million persons) from radon and radon daughter products as a result of repository mining operations ranges from less than one to 100 man-rem depending on host rock. During the time the repository was receiving wastes (6 to 20 years), normal operations might add about 1 man-rem to this total. During these time periods, the regional population would have received from about 1,000,000 to 4,000,000 man-rem from naturally occurring, undisturbed radionuclides. Thus, construction and operation of a geologic repository under normal conditions do not constitute a significant radiological impact.

Accidents occurring during operation of the repository that might have radiological impacts were also investigated. The accident believed to have the largest potential radiological consequence is the dropping of a waste canister down the repository shaft and rupture of the canister on impact. The 70-year whole-body doses to the regional population from such accidents were determined to total to less than 6000 man-rem for 20 years of waste emplacement in a repository. During the same period the regional population would receive about 4,000,000 man-rem from naturally occurring sources. However, doses to workers in the repository from radioactive material released in the event of a canister drop could be fatal (greater than 7,000 rem in first year following the accident). Engineered precautions similar to those outlined in Section 5.4 are expected to preclude such consequences and to reduce doses to workers to safe levels.

Results of a total system analysis of radiological and other impacts for the various power generating projections are summarized in Section 1.6. For those interested in details of environmental aspects of the complex interactions of predisposal and disposal activities, and power growth assumptions, Chapter 7 should be consulted.

1.3.3.2 Resource Commitments

Various resources would be required in the construction and operation of geologic repositories. Ranges of some of the more important resource commitments, as a function of host rock, are presented in Table 1.3.2. The values given are based on a normalized energy production basis of one GWe-yr (about 9 billion kWh, equivalent to one large reactor operating for one year).

Even at an installed nuclear power capacity of 250 GWe operating over several decades the tabulated material and energy commitments are but a small fraction of that used for the

TABLE 1.3.2 Resource Commitments Associated with Construction and Operation of Geologic Waste Repositories, Normalized to 1 GWe-yr

	Spent Fuel Repositories	Fuel Reprocessing Waste Repositories	Approximate U.S. Annual Production
Propane, m^3	1.6 - 2.0	1.5 - 3.3	1×10^6
Diesel Fuel, m^3	$1.2 \times 10^2 - 1.7 \times 10^2$	$1.7 \times 10^2 - 2.5 \times 10^2$	4×10^8
Gasoline, m^3	$1.2 \times 10^1 - 1.5 \times 10^1$	$1.1 \times 10^1 - 2.4 \times 10^1$	6×10^8
Electricity, kw-hrs	$1.0 \times 10^6 - 1.1 \times 10^6$	$1.3 \times 10^6 - 1.8 \times 10^6$	2×10^{12}
Manpower, man-yrs	$1.6 \times 10^1 - 1.7 \times 10^1$	$1.8 \times 10^4 - 3.3 \times 10^1$	4×10^6 (a)
Steel, MT	$2.5 \times 10^1 - 6.1 \times 10^1$	$6.2 \times 10^1 - 1.0 \times 10^2$	1×10^8
Cement, MT	$2.2 \times 10^1 - 2.6 \times 10^1$	$2.9 \times 10^1 - 6.7 \times 10^1$	7×10^7
Lumber, m^3	1.7 - 2.1	1.6 - 3.5	3×10^9

(a) Construction and mining.

total economy. To give additional perspective to the consumption of energy as fossil fuel and electricity, each was converted to units of energy expended in deep geologic disposal of waste per unit of energy produced by the fuel from which the waste came. In the case of spent fuel 0.04% of the energy produced was consumed in geologic waste disposal and in the case of fuel reprocessing wastes 0.05% of the energy produced was consumed. On this basis it is concluded that the irretrievable commitment of the above materials is warranted.

1.3.3.3 Socioeconomic Impacts

Socioeconomic impacts associated with the construction and operation of repositories are dependent largely on the number of persons who move into the locality in which the facility will be located. Site characteristics that are especially important in influencing the size of the impacts include the availability of a skilled local labor force, secondary employment, proximity to a metropolitan area, and demographic diversity (population size and degree of urbanization) of counties in the commuting region. An additional factor in the generation of impacts is the time pattern of project-associated population change. For example, a large labor force buildup followed closely by rapidly declining project employment demand could cause serious economic and social disruptions both near the site and within the commuting region.

In this Statement impacts are estimated for three reference sites, identified as Southeast, Midwest, and Southwest. These areas were chosen because siting of facilities in those regions is plausible and because they differ substantially in demographic characteristics, thus providing a reasonable range of socioeconomic impacts.

In general, the reference Southwest site is more likely to sustain significant socioeconomic impacts than are the other two sites, because it has a smaller available unemployed construction labor force, lacks a nearby metropolitan center, and is subject to the generation of greater secondary employment growth than are the other sites. If a repository were to be built in an area where demographic conditions approximated those of the Southwest site, a detailed analysis of site-specific socioeconomic impacts would be needed to help prevent serious disruptions in provision of necessary social services.

Table 1.3.3 presents the manpower requirements for construction and operation of a single waste repository accepting either spent fuel or reprocessing wastes.

TABLE 1.3.3. Manpower Requirements for Construction and Operation of a Single Waste Repository (three peak years)

Repository Medium	Average Annual Employment			
	Spent Fuel Repository		Reprocessing Waste Repository	
	Construction	Operation	Construction	Operation
Salt	1700	870	2000	1300
Granite	4200	1100	3000	1300
Shale	2200	880	2100	1200
Basalt	5000	1100	3800	1500

1.3.3.4 Land Use, Ecological Impacts and Other Impacts

At an 800-ha repository, above ground facilities (including mining spoils piles) would occupy about 200 to 300 ha depending on geologic media. An additional 10 ha would be used for access roads. An 800-ha area above the subterranean repository would be set aside at the surface, and mineral and surface rights would be restricted. This surface land, except that occupied by mining spoils piles, could be returned to its former use when the repository surface facilities are decommissioned after sealing and closure of the repository. Presently an area equal to 3,200 ha, centered over the repository, is considered necessary for exclusion of nearby subsurface activities. Subsurface activities could be restricted as long as institutional control exists. (It is expected that this issue will be more closely examined for site-specific applications. Present plans call for a repository design that does need not to rely on institutional controls after closure.)

The main ecological concern of repository construction and operation is the potential for airborne and waterborne contamination of the environs as a result of the very large mine spoils piles. Land near repositories in salt could be contaminated by windblown salt;

nearby streams could be harmed by runoff contaminated with salt. Removal of the salt to a nonharmful environment, such as through dilute dispersal at sea or stabilization of the salt piles could obviate the problem. Repositories in shale do not appear to pose as serious a problem, although alteration of pyrite, a mineral found in shales, could lead to contamination of streams. The spoils piles from repositories in granite and basalt are not expected to have a significantly adverse affect on the environment.

It is possible that for any rock type the pile of rock left on the surface will have an adverse aesthetic impact. The possibility also exists that these spoils piles of rock (millions of MT), if arranged properly, could become markers identifying the locations of the repositories--although some would maintain that such markers eventually might actually enhance the probability of archaeological exploration.

It is concluded that, in a generic sense, neither land use nor ecological impacts are of such a magnitude as to deter development of geologic repositories or their use for disposal of nuclear radioactive wastes from commercial power generation.

1.3.4 Environmental Impacts in the Long Term

Planned functioning of the geologic repository after closure will result in very little in the way of environmental impacts. So long as institutional controls exist there will probably be some control of land useage above the repository. There will probably be some monitoring performed until future generations decide to discontinue monitoring. Although heat from the waste will ultimately reach the surface over the repository, the estimated temperature rise is expected to be less than 0.5°C in all cases. Small amounts of uplift and subsidence might occur for repositories in salt and shale but probably none for repositories in granite and basalt. During planned functioning of the waste repository after closure there will be no health effects attributable to the repository.

Although waste repositories will be sited, loaded, and sealed with every expectation that long term radiological impacts will be nonexistent, the ways in which a repository might fail, the likelihood of its failure, and the consequences to the human environment of such failure were investigated in detail. At 600 m below the earth's surface, it is extremely improbable that wastes in biologically important concentrations would ever reach the human environment. Nevertheless, several events were postulated that might release repository contents, and estimates were made of the possible consequences of such release, in terms of radiation dose to, and postulated health effects among, the public. In brief, these events were:

- impact of a giant meteorite directly over the repository releasing some of the repository contents to the atmosphere (which is believed to have consequences on the order of other events such as volcanism and nuclear warfare that might breach a repository)
- faulting or other fracturing of the host rock, followed by flooding of the repository with water and either a) contamination of an emergent stream, b) slow ground-

water transport to the biosphere, or c) contamination of a near surface aquifer that had been tapped by a well

- human intrusion by drilling for exploration
- solution mining of salt in the case of a repository in salt.

The doses to the regional population were calculated for each event and then the number of radiation-related health effects was determined by applying a conversion factor of from 100 to 800 health effects (50 to 500 fatal cancers plus 50 to 300 serious genetic disorders) per million man-rem (as developed in Appendix E). The results were then multiplied by the probability (where determinable) that the event would occur, to obtain a measure of expected societal risk.

Societal risk in each case where probabilities could be estimated were very small; for example, in the case of breach by a giant meteorite whose probability was estimated to be 2×10^{-13}/yr and where the largest calculated consequences were 1.4×10^5 health effects, the societal risk amounted to 3×10^{-8} health effects/yr, and in the case of faulting and flooding the societal risk amounted to 3×10^{-11} health effects/yr. For comparison, the expected societal risk from lightning in the population of 2 million, in the reference environment, is about 1 fatality per year. In the worst case of general contamination of water, not more than one radiation-related fatality was projected to result over a 10,000-year period.

Although believed to be highly unlikely because of the extreme depth of the repository, no probability could be assigned to the act of drilling into a repository. If, however, drilling did take place within the surface projection of the repository area and to the depth of the repository, the probability was determined to be 0.005 per 1000 drill holes (based on relative cross-sections and spatial density of canisters in the repository) that a waste canister would be intercepted. If drilling took place about 1000 yrs after disposal and a high-level waste canister were penetrated, the contaminated drilling mud, when brought to the surface, could result in a small increase in risk of adverse health effects occuring among about two dozen people postulated to live in the immediate area, if no cleanup takes place.

Even if drilling into the repository were to occur without canister penetration the drill hole might constitute a conduit for entry of water into the repository. Mechanisms to return the water to the biosphere are more difficult to postulate. Regardless, if this event took place, the consequences are believed to be significantly less than those resulting from faulting and flooding scenarios also discussed in this Statement.

Because of the abundance of salt in this country, and its frequent location at depths much less than 600 m, the chance of solution mining near a repository in bedded salt formations is believed to be remote. However, solution mining in a domed salt formation is

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(a) The production rate of the hypothetical salt solution mine was estimated to be sufficient to supply salt for about 40 million people.

believed to be much more likely. Part of the reason for this is that there may be geologic surface features that suggest the presence of domed salt; however these features are absent for deeply bedded salt. Assuming that a repository in salt was breached in the course of solution mining for salt and that salt was mined for one year before it was discovered to be contaminated, doses about one-tenth of those from naturally occurring sources were calculated to result among the 40 million people assumed to be consuming the contaminated salt.[a] Health effects were also estimated to be about one-tenth of those that might be attributable from natural background.

1.4 ALTERNATIVE ACTION--BALANCED DEVELOPMENT OF ALTERNATIVE DISPOSAL METHODS[(a)]

The alternative program strategy calling for balanced development of several alternative methods requires selection of some other disposal alternative(s) in addition to mined geologic repositories. The following disposal methods are analyzed as candidates for consideration in the alternative waste disposal program. and from this analysis, mined geologic, very deep hole, and subseabed disposal are identified as the most likely candidate technologies for balanced development.

1.4.1 Very Deep Hole Waste Disposal Concept

A very deep hole concept has been suggested that involves the placement of nuclear waste in holes in geologic formations as much as 10,000 meters (6 miles) underground. Potential rock types for a repository of this kind include crystalline and sedimentary rocks located in areas of tectonic and seismic stability.

Spent fuel or high-level waste canisters could be disposed of in very deep holes. However, it is not economically feasible to dispose of high-volume wastes (e.g., TRU) in this manner and thus another alternative, such as deep geologic repositories, is also required if spent fuel is reprocessed. There is some question whether or not drilling of holes to the depths suggested and in the sizes required can be achieved.

The principal advantage of the very deep hole concept is that certain (but not all) wastes can be placed farther from the biosphere, in a location where it is believed that circulating ground water is unlikely to communicate with the biosphere.

1.4.2 Rock Melt Waste Disposal Concept

The rock melt concept for radioactive waste disposal calls for the direct placement of liquids or slurries of high-level wastes or dissolved spent fuel, with the possible addition of small quantities of other wastes, into underground cavities. After the water has evaporated, the heat from radioactive decay would melt the surrounding rock. The melted rock has been postulated to form a complex waste form by reaction with the high-level waste. In about 1000 years, the waste-rock mixture would resolidify, trapping the radioactive material in what is believed to be a relatively insoluble matrix deep underground. Since solidification takes about 1000 years the waste is most mobile during the period of greatest fission product hazard.

Not believed to be suitable for rock melt disposal are wastes from reprocessing activities such as hulls, end fittings, and TRU wastes remaining after dissolution. Because of the inability to accommodate these wastes, some other disposal method would have to be used in conjunction with the rock melt disposal concept.

(a) Analyses developed in this Statement under the alternative program evaluate the environmental impacts of deferring implementation of a disposal program until the year 2030. This situation can also be interpreted as demonstrating impacts that would result from a delayed disposal program.

1.4.3 Island-based Geologic Disposal Concept

Island-based disposal involves the emplacement of wastes within deep stable geological formations, much as in the conventionally mined geologic disposal concept and in addition relies on a unique hydrological system associated with island geology. Island-based disposal would accommodate all forms of waste as would conventionally mined geologic disposal; however, additional port facilities and additional transportation steps would be required. Remoteness of the probable candidate islands has been cited as an advantage in terms of isolation.

1.4.4. Subseabed Disposal Concept

It has been suggested that wastes could be isolated from the biosphere by emplacement in sedimentary deposits beneath the bottom of the deep sea (thousands of meters below the surface), which have been deposited over millions of years. The deposits have been shown by laboratory experiments to have high sorptive capacity for many radionuclides that might leach from breached waste packages. The water column is not considered a barrier, however it will inhibit human intrusion and can contribute to dilution by dispersal of radionuclides that might escape the sediments.

One subseabed disposal system incorporates the emplacement of appropriately treated waste or spent reactor fuel in free-fall needle-shaped "penetrometers" that, when dropped through the ocean, would penetrate about 50 to 100 m into the sediments. A ship designed for waste transport and placement would transport waste from a port facility to the disposal site and would be equipped to emplace the waste containers in the sediment.

Subseabed disposal is an attractive alternative disposal technique because technically it appears feasible that, at least for high-level waste and spent fuel, the waste can be placed in areas having relatively high assurance of stability. If at some point in time all of the barriers failed, the great dilution and slow movement should retard the return of radionuclides to the human environment in biologically important concentrations. The research needed to technically permit subseabed disposal to go forward has been projected not to be as costly or time consuming as some other alternatives. On the other hand, like island-based geologic disposal, the subseabed concept has the disadvantage of the need for special port facilities and for additional transportation steps in comparison to mined repositories on the continent.

As noted, subseabed disposal is believed to be technologically feasible; however, international and domestic legal problems to its implementation would require favorable resolution. Whether subseabed disposal can provide isolation of wastes equal to that of deep geologic repositories has not been fully assessed. Because of volume considerations, subseabed disposal does not appear practical for TRU wastes and some other method would be required for their disposal.[a]

(a) Trenches in the ocean floor have been suggested as a means of disposing of higher volume, but less radioactive wastes.

1.4.5 Ice Sheet Disposal Concept

Disposal in continental ice sheets has been suggested as a means of isolating high-level radioactive waste. Past studies have specifically addressed the emplacement of waste in either Antarctica or Greenland. The alleged advantages of ice sheet disposal, which are disposal in a cold, remote area and in a medium that should isolate the wastes from man for many thousands of years, cannot be proven on the basis of current knowledge.

Proposals for ice sheet disposal of high-level waste and/or spent fuel suggest three emplacement concepts:[a]

- Passive slow descent--waste is emplaced in a shallow hole and the waste canister melts its own way to the bottom of the ice sheet
- Anchored emplacement--similar to passive slow descent but an anchored cable limits the descent depth and allows retrieval of the canister and prevents movement to the bottom of the sheet.
- Surface storage--storage facility supported above the ice sheet surface with eventual slow melting into the sheet.

Ice sheet disposal, regardless of the emplacement concept, would have the advantages of remoteness, low temperatures, and isolating effects of the ice. On the other hand, transportation and operational costs would be high, ice dynamics are uncertain, and adverse global climatic effects as a result of melting of portions of the ice are a remote possibility. The Antarctic Treaty now precludes waste disposal in the Antarctic ice sheet. The availability of the Greenland ice sheet for waste disposal would depend upon acceptance by Denmark and the local government of the island itself.

A great deal of research appears to be needed before the potential of ice sheet disposal is determined. Even though the apparent bowl-shaped ice cap of Greenland would result in the wastes melting to the bottom of the bowl where they might remain permanently, the consequences of release of radioactive decay heat to the ice are uncertain. Because of weather extremes and environmental conditions on the ice sheets, difficulties are also predicted for transportation of the wastes to the site, waste emplacement and site characterization.

1.4.6 Well Injection Disposal Concepts

Two methods of well injection have been suggested: deep well liquid injection and shale/grout injection.

Deep well liquid injection involves pumping acidic liquid waste to depths of 1000 to 5000 m (3,300 to 16,000 ft) into porous or fractured strata that are suitably isolated from the biosphere by relatively impermeable overlying strata. The waste is expected to remain

(a) Present concepts for waste disposal in ice sheets call for TRU reprocessing waste to be placed in mined geologic waste repositories.

in liquid form and may thus progressively disperse and diffuse throughout the host rock. Unless limits of movement are well defined, this mobility within the porous host media formation would be of concern regarding eventual release to the biosphere.

For the shale/grout injection alternative, the shale is fractured by high-pressure injection and then the waste, mixed with cement and clays, is injected into the fractured shale formations at depths of 300 to 500 m (1000 to 1600 ft) and allowed to solidify in place in a set of thin solid disks. Shale has very low permeability and predictably good sorption properties. The formations selected for injection would be those in which it can be shown that fractures would be created parallel to the bedding planes and in which the wastes would be expected to remain within the host shale bed. This requirement is expected to limit the injection depths to the range stated above.

This alternative is applicable only to reprocessing wastes or to spent fuel that has been processed to liquid or slurry form. Therefore, well injection is not sufficient to dispose of all wastes generated, and a suitable additional technique would be required.

1.4.7 Transmutation Concept

In the reference transmutation concept, spent fuel would be reprocessed to recover uranium and plutonium (or processed to obtain a liquid high-level waste stream in the case where uranium and plutonium are not to be recycled). The remaining high-level waste stream is partitioned into an actinide waste stream and a fission product stream. The fission product stream is concentrated, solidified, and sent to a mined geologic repository for disposal. The waste actinide stream is combined with uranium or uranium and plutonium, fabricated into fuel rods, and reinserted into a reactor. In the reactor, about 5 to 7% of the recycled waste actinides are transmuted to stable or short-lived isotopes, which are separated out during the next recycle step for disposal in the repository. Numerous recycles would result in nearly complete transmutation of the waste actinides; however, additional waste streams are generated with every recycle. Transmutation, however, provides no reduction in the quantities of long-lived fission product radionuclides such as ^{99}Tc and ^{129}I in the fission product stream that is sent to geologic disposal.

1.4.8 Space Disposal Concept

Space disposal has been suggested as a unique option for permanently removing high-level nuclear wastes from the earth's environment. In the reference concept, high-level waste is formed into a ceramic-metal matrix, and packaged in special flight containers for insertion into a solar orbit, where it would be expected to remain for at least one million years. The National Aeronautics and Space Administration (NASA) has studied several space disposal options since the early 1970s. The concept involves the use of a special space shuttle that would carry the waste package to a low-earth orbit where a transfer vehicle would separate from the shuttle and place the waste package and another propulsion stage into an earth escape trajectory. The transfer vehicle would return to the shuttle while the remaining rocket stage inserts the waste into a solar orbit.

Space disposal is of interest because once the waste is placed in orbit its potential for environmental impacts and human health effects is judged to be nonexistent. However, the risk of launch pad accidents and low earth orbit failures have not been determined.

The space disposal option appears feasible for selected long-lived waste fractions of radionuclides such as ^{129}I, or even for the total amount of reprocessed high-level waste that will be produced. Space disposal of unreprocessed fuel rods and other high volume wastes does not appear economically feasible or practical because of the large number of flights involved.

1.5 NO-ACTION ALTERNATIVE

The no-action alternative would leave spent fuel or reprocessing wastes at the sites generating the waste or possibly at other surface or near-surface storage facilities for an indefinite time. In this alternative, existing storage is known to be temporary and no consideration has been given to the need for additional temporary storage when facilities in use have exceeded their design lifetime. There seems to be no question but that at some point in time wastes will require disposal and that considerable time and effort will be required to settle upon an adequate means of disposal. It seems clear that development of acceptable means of disposal of wastes is sufficiently complex and of sufficiently broad national importance that coordination of research and development, construction, operation, and regulation at the Federal level is required and that the no-action alternative is unacceptable. Indeed, adoption of a no-action alternative by the Department of Energy could be construed as not permissible under the responsibility mandated to the Department by law. Neither would a no-action alternative be in accord with the President's message of February 12, 1980, when he stated that "...resolving...civilian waste management problems shall not be deferred to future generations."

1.6 PREDISPOSAL SYSTEMS[a]

After the wastes are generated and before they are disposed of, several predisposal operations are required. The combination of these operations is referred to as a predisposal system. System operations include treatment and packaging to prepare the waste for the specific requirements of a disposal option, interim storage if the treated waste cannot be shipped immediately to a disposal site, shipment to interim storage and/or to a disposal site, and decommissioning of the waste treatment and storage facilities. In considering various alternatives for disposal of wastes, different operations for predisposal treatment required by each alternative must also be compared.

All of the alternatives that utilize a dissolution process would also generate considerable quantities of miscellaneous TRU waste. It is assumed here that these materials are always sent to a mined geologic repository regardless of the disposal option selected for high-level waste.

1.6.1 Predisposal System for the Once-Through Cycle

Following discharge from the reactor, spent fuel is stored for a period of time at reactor storage basins. The fuel is then shipped to a treatment and/or packaging facility if a disposal facility is available. If a disposal facility is not available at the end of the reactor storage period, the fuel is assumed to be shipped to an away-from-reactor (AFR) storage facility and subsequently shipped to available repositories. When a disposal facility is available at the end of the reactor storage period, the fuel is shipped to a treatment and/or packaging facility. If the disposal site is separate from the treatment and/or packaging facility, the fuel is then shipped to the disposal site.

Initial storage and shipment operations are identical for all of the disposal alternatives. The differences imposed on the predisposal systems by the disposal alternatives are in the treatment and/or packaging and final shipment to disposal.

1.6.2 Predisposal System for the Reprocessing Cycle

In the reprocessing cycle, wastes requiring disposal are produced at the fuel reprocessing plant (FRP) and at the mixed-oxide fuel fabrication plant (MOX-FFP). Both high-level waste and TRU waste are produced at the FRP but only TRU wastes are produced at the MOX-FFP. These wastes are assumed to be treated and packaged at the site where they are produced, either the FRP or MOX-FFP. They are then shipped to interim storage if a disposal facility is not available; finally, they are shipped to a disposal facility.

1.6.3 Accident Impact Summary for Predisposal Operations

Table 1.6.1 summarizes the results of the predisposal-system accident analyses. This table shows that transportation is the waste management step with the potential for the

(a) Although this section is very brief, predisposal systems involve many facilities, operations, and processes and for those interested, details are given in Chapter 4.

TABLE 1.6.1. Summary of Radiation Effects from Potential Worst-Case Predisposal System Accidents

	70-Year Dose to Maximum-Exposed Individual, rem	
	Once-Through Cycle	Reprocessing Cycle
Transportation (impact and fire)		
Spent Fuel (4-year-old)	0.6[a]	
HLW		10[b]
TRU Waste		3
Storage	5×10^{-2}	8×10^{-3}
Treatment and Packaging	3×10^{-5}	2×10^{-3}

(a) Shipment of 6-month-old spent fuel, which is unlikely, could result in a maximum individual dose of 130 rem.
(b) The age of HLW at shipment in the scenario used in this Statement would be about 6-1/2 years old.

most serious accidents in either fuel cycle. The estimated exposures in these accidents, however, are not large enough to cause observable clinical effects. Only in the case of an accident involving shipment of 6-month-old fuel was the dose (130 rem) determined to be sufficiently large that the individuals exposed would have a significant increase in probability of developing cancer sometime during their life or of passing on a genetic defect.

1.7 ENVIRONMENTAL IMPACTS OF PROGRAMMATIC ALTERNATIVES FOR THE ONCE-THROUGH AND THE REPROCESSING FUEL CYCLE OPTIONS AND VARIOUS NUCLEAR POWER GROWTH ASSUMPTIONS

To assess and compare the overall impacts of implementing the three programmatic alternatives addressed in this Statement, an analysis was made using a computer simulation of the complete waste management system functioning over the entire post-fission lifetime of a nuclear power system. This analysis considers treatment and disposal of all post-fission high-level wastes (spent fuel or reprocessing HLW), airborne wastes[a] and transuranic (TRU) wastes including decommissioning wastes. In this analysis all waste management functions are accounted for and all radioactive waste streams are tracked each year from origin through treatment, storage, transport and accumulation in a disposal repository.

Both the once-through cycle and the reprocessing cycle are addressed for the proposed and alternative programmatic actions for the nuclear power scenarios presented in Table 1.1.1. For the no-action alternative, indefinite storage of spent fuel in water basin facilities with no ultimate disposal was assumed and reprocessing is not considered. Only the first three nuclear growth cases are considered for the no-action alternative, because, without disposal, growth of nuclear power beyond year 2000 does not appear credible.

DOE estimates that implementation of the proposed program will result in the establishment of operating geologic repositories within the time range of 1997 to 2006. An exact date of operation, depending on a number of variables, will be determined by the outcome of existing programs. To cover additional contingencies such as an accelerated effort to open a repository or, at the other extreme, additional delays for reasons not yet foreseen, a range of repository startup dates from 1990 to 2010 is considered here. The range of impacts is important in this simulation rather than the specific dates of repository startup.

Implementation of the alternative program would result in extending the time to operation of the first disposal system. This action implies a further period of research and development to bring the development status of the selected disposal alternatives to an approximately equal status with current knowledge regarding geologic disposal. At that time, a preferred technology would be selected and effort would be concentrated on developing this preferred technology with a program similar to the currently planned program for implementing geologic disposal. Thus a substantial time delay is inherent in this alternative. Implementation of this alternative program is simulated by a range of repository startup dates from 2010 to 2030.

In the system analysis, mined geologic repositories are used to simulate the disposal method ultimately selected under the alternative program. (This concept is the only one developed sufficiently to model impacts and costs reasonably well, and any alternative disposal concept that might be selected would only be selected if it did not have significantly greater impacts or costs.) The principal effects of the alternative program implementation are the required interim storage for spent fuel or reprocessing wastes, the additional

(a) Airborne wastes from nuclear power plants are not considered in this Statement because such wastes are considered in the EIS prepared for each nuclear power plant.

transportation to and from this storage and the impacts and costs for these operations. Benefits of the delay inherent in this alternative program include the processing and disposal of older and thus less radioactive and cooler wastes.

Repository startup dates considered in the once-through cycle and reprocessing cycle system simulations are shown in Tables 1.7.1 and 1.7.2, respectively. The range of reprocessing startup dates considered is also shown in Table 1.7.2. To simplify the analysis only a single mid-range repository startup date, year 2000 representing the proposed program and 2020 representing the alternative program, was used for Cases 4 and 5. For the same reason only a single mid-range reprocessing date was used for these cases. However, the same potential range as in the other cases should be inferred for both repositories and reprocessing.

TABLE 1.7.1. Repository Startup Dates Considered in the Once-Through-Cycle System Simulations

Nuclear Power Growth Cases	Proposed Program	Alternative Program	No-Action Alternative
1. Present Inventory Only	1990 to 2010[a]	2010[a] to 2030	None
2. Present Capacity and Normal Life	1990 to 2010[a]	2010[a] to 2030	None
3. 250 GWe System by Year 2000 and Normal Life	1990 to 2010[a]	2010[a] to 2030	None
4. 250 GWe System by Year 2000 and Steady State	2000	2020	--
5. 500 GWe System by Year 2040	2000	2020	--

(a) These cases are the same under both the proposed and alternative programs.

TABLE 1.7.2. Reprocessing and Repository Startup Date Combinations Considered in the Reprocessing-Cycle System Simulations

Nuclear Power Growth Cases	Proposed Program		Alternative Program	
	Reprocessing	Repository	Reprocessing	Repository
1. Present Inventory	NA[a]	NA[a]	NA	NA
2. Present Capacity and Normal Life	NA	NA	NA	NA
3. 250 GWe System by Year 2000 and Normal Life	1990 1990 2010	1990[b] 2010[b] 2010[b]	1990 2010 1990 2010	2010[b] 2010[b] 2030 2030
4. 250 GWe System by Year 2000 and Steady State	2000	2000	2000	2020
5. 500 GWe System by Year 2040	2000	2000	2000	2020

(a) NA = not applicable. Reprocessing assumed not to be undertaken in these low-growth cases.
(b) These cases are the same under both the proposed and alternative programs.

1.7.1 System Radiological Impacts

Both the regional (reference environment of 2 million persons) and worldwide 70-year whole-body dose accumulations for the proposed program, the alternative program, and the no-action alternative are compared for the once-through cycle in Table 1.7.3. Somewhat higher dose accumulations are indicated for the alternative program than for the proposed program. However, the differences are not large enough to be significant.[a] The dose accumulation for the no-action alternative is somewhat less than for the other alternatives, but considering the time period involved, the differences are not significant. As would be expected, the dose increases with increasing size of the nuclear systems served.

TABLE 1.7.3. Comparison of 70-Year Whole-Body Dose Accumulations from Normal Operations for the Program Alternatives Using the Once-Through Cycle, man-rem

Case	Nuclear Power Growth Assumption	Proposed Program (Geologic Disposal Starting 1990 - 2010) Regional	Worldwide	Alternative Program (Disposal Starting 2010 - 2030) Regional	Worldwide	No-Action Alternative Regional	Worldwide
1	Present Inventory Only	36	48	36	48	0.2	4
2	Present Capacity Normal Life	200 to 250	290 to 370	250 to 260	370 to 380	90	160
3	250 GWe System by Year 2000 and Normal Life	940 to 1200	1400 to 1800	1200 to 1300	1800 to 1900	480	800
4	250 GWe System by Year 2000 and Steady State	1400	2100	1800	2600	NA[a]	NA
5	500 GWe system by Year 2040	1900	2800	2400	3400	NA	NA
	Dose Accumulation from Natural Radiation Sources	1×10^7	4.5×10^{10}	1×10^7	4.5×10^{10}	1×10^7	4.5×10^{10}

(a) NA = not applicable.

The regional and worldwide 70-year whole-body dose accumulations for the proposed and alternative programs are compared for the reprocessing case in Table 1.7.4. The doses are much larger here than in the once-through cycle. However, the dose from reprocessing is only a small fraction of the naturally occurring dose even in the highest nuclear growth case examined here; i.e., 0.5% of the regional dose and 0.003% of the worldwide dose. The doses from either the proposed program or the alternative program are the same. The regional and worldwide dose is accumulated principally (about 95%) from the waste treatment operations and the same quantities of waste are treated in either alternative--the only difference is that waste production and treatment occur at different times.

(a) Result in less than one additional health effect as will be shown in following tables.

TABLE 1.7.4. Comparison of 70-Year Whole-Body Dose Accumulations from Normal Operations for the Program Alternatives Using the Reprocessing Cycle,[(a)] man-rem

Case	Nuclear Power Growth Assumption	Proposed Program (Geologic Disposal) Starting 1990 - 2010) Regional	Worldwide	Alternative Program (Disposal Starting 2010 - 2030) Regional	Worldwide	No-Action Alternative Regional	Worldwide
1	Present Inventory Only	NA[(b)]	NA	NA	NA	NA	NA
2	Present Capacity and Normal Life	NA	NA	NA	NA	NA	NA
3	250 GWe System by Year 2000 and Normal Life	13,000 to 33,000	580,000 to 970,000	13,800 to 33,000	580,000 to 970,000	NA	NA
4	250 GWe System by Year 200 and Steady State	33,000	1,000,000	33,000	1,000,000	NA	NA
5	500 GWe System by Year 2040	46,000	1,500,000	46,000	1,500,000	NA	NA
	Dose Accumulation from Natural Radiation Sources	1×10^7	4.5×10^{10}	1×10^7	4.5×10^{10}	1×10^7	4.5×10^{10}

(a) Assumed reprocessing startup dates range from 1990 to 2000.
(b) NA = not applicable.

In this Statement, 100 to 800 health effects (50 to 500 total cancers plus 50 to 300 serious genetic disorders) are postulated to occur in the exposed population per million man-rem. Based on this criterion, the program alternatives are compared on the basis of health effects in Table 1.7.5 for the once-through cycle and Table 1.7.6 for the reprocessing cycle.

For the once-through cycle, with the high nuclear growth assumption, the number of health effects range from 0 to 2 on a regional basis and 0 to 3 on a worldwide basis. In the reprocessing case, the number of health effects are larger. For the high nuclear growth assumption, they range from 5 to 37 health effects on a regional basis and from 140 to 1100 on a worldwide basis. However, the health effects calculated to occur over the same period from naturally occurring radioactive sources range from 1000 to 8000 health effects to the regional population and 4×10^6 to 4×10^7 health effects to the worldwide population.

1.7.2 System Resource Commitments

Estimates of major resource commitments for construction and operation of the entire waste management system were developed for each of the nuclear growth assumptions and each repository and reprocessing startup date. The resources considered include steel, cement, diesel fuel, gasoline, propane, electricity and manpower.

TABLE 1.7.5. Comparison of Health Effects for the Program Alternatives Using the Once-Through Cycle

Case	Nuclear Power Growth Assumption	Number of Effects: Proposed Program (Geologic Disposal) Starting 1990 - 2010) Regional	Worldwide	Alternative Program (Disposal Starting 2010 - 2030) Regional	Worldwide	No-Action Alternative Regional	Worldwide
1	Present Inventory Only	0	0	0	0	0	0
2	Present Capacity and Normal Life	0	0	0	0	0	0
3	250 GWe System by Year 2000 and Normal Life	0 to 1	0 to 2	0 to 1	0 to 2	0	0 to 1
4	250 GWe System by Year 2000 and Steady State	0 to 1	0 to 2	0 to 1	0 to 2	NA[(a)]	NA
5	500 GWe System by Year 2040	0 to 2	0 to 2	0 to 2	0 to 3	NA	NA

(a) NA = not applicable.

TABLE 1.7.6 Comparison of Health Effects for the Program Alternatives Using the Reprocessing Cycle

Case	Nuclear Power Growth Assumption	Number of Effects: Proposed Program (Geologic Disposal Starting 1990 - 2010) Regional	Worldwide	Alternative Program (Disposal Starting 2010 - 2030) Regional	Worldwide	No-Action Alternative Regional	Worldwide
1	Present Inventory Only	NA[(a)]	NA	NA	NA	NA	NA
2	Present Capacity and Normal Life	NA	NA	NA	NA	NA	NA
3	250 GWe System by Year 2000 and Normal Life	1 to 26	6 to 750	1 to 26	6 to 750	NA	NA
4	500 GWe System by Year 2040	3 to 27	100 to 800	3 to 27	100 to 800	NA	NA
5	500 GWe System by Year 2040	5 to 37	140 to 1100	5 to 37	140 to 1100	NA	NA

(a) NA = not applicable.

For the proposed program, resource requirements for reprocessing are somewhat higher than for the once-through cycle in the case of steel, cement, electricity, and manpower; are about the same to somewhat higher for diesel fuel and gasoline; and are substantially higher for propane. The higher propane requirement results from incineration of combustible waste. Gasoline and diesel fuel are used primarily in transportation. These fuel requirements are based on present practice and can be expected to change as fuel use patterns change generally. The propane requirements for the reprocessing cycle represent about 0.5% of the total U.S. consumption for the period to year 2050 assuming current consumption rates hold constant. The largest diesel fuel use amounts to about 1% of total U.S. consumption over the period.[a] Electricity consumption amounts to 0.02 to 0.05% to the total energy generated by the nuclear power system in this case.

The resource commitments for the program alternatives using the once-through cycle increase as the size of the nuclear system served increases. With the exception of the present inventory case which changes only slightly, requirements for the alternative program compared to the proposed program tend to range up to 2 to 3 times higher for steel, cement, gasoline, propane, and manpower and modestly higher for diesel fuel and electricity. Requirements for the no-action alternative are zero in the present inventory case and are about the same as the alternative program for steel, cement, gasoline, propane, and manpower but diesel and electricity consumption are much lower.

Resource commitments for the program alternatives in the reprocessing cycle tend to be about the same to somewhat higher than for the proposed program requirements.

1.7.3 Systems Costs[b]

Both total cost and levelized[c] unit costs (per kWh) were developed. These costs include all waste treatment, storage, transport and disposal costs for wastes resulting from nuclear power generation through the year 2040. The costs also include DOE's research and development and repository site qualification costs which are assumed to be recovered through fees charged to the utilities for storage and disposal. The cost ranges consider four different disposal media.

In terms of total costs, the costs increase with increasing size of the nuclear system but are disproportionately high for the very low-growth cases. The estimated costs range from \$5 to \$12 billion for the present inventory case (Case 1), to \$80 to \$150 billion for the system that reaches 500 GWe installed capacity in the year 2040 (Case 5). Of these totals, the estimated R&D and multiple-site qualification costs range from \$2.9 to \$3.6 billion at the low end of the proposed program to \$9 to \$10 billion at the high end of the

(a) While a commitment of 1% of current U.S. consumption may appear small, some commenters on the draft Statement viewed such a quantity as excessively large in terms of commitment for a single industrial use. It should be noted that resource needs have been approximated for this final Statement. It is believed that optimizing, for instance in terms of shipping distances, could result in reduction of quantities of resource required.

(b) All costs are cited in terms of 1978 dollars.

(c) Levelized Unit Cost = $\frac{\text{Annualized Capital and Operating Costs}}{\text{Annualized Units Produced}}$.

alternative program. The range of costs for the alternative program is higher than the proposed program for the once-through cycle but about the same for the reprocessing cycle. Costs for the no-action alternative are about the same as the low end of the range for the proposed program.

The costs can be better placed in perspective when shown as unit costs per kWh of generated electrical energy. The levelized unit costs are sensitive to the discount rate used (cost of money). Because waste management costs are incurred after the generation of the electricity, increasing the discount rate has the effect of reducing the unit cost. A range of discount rates from 0 to 10% is considered in this Statement and a 7% rate was selected for illustration in this summary. Since the unit cost for the once-through cycle and the reprocessing cycle are similar, the unit costs for the program alternatives are compared in Table 1.7.7 without distinguishing the cost range for each fuel cycle. Costs are somewhat higher when a 0% discount rate is used and slightly lower with a 10% discount rate. On this basis there is little difference between the proposed program and alternative program costs. Cost of electricity in 1978 averaged 3.5 ¢/kWh over all types of services throughout the U.S. On that basis the additional cost for waste management and disposal would add about 2 to 6% to the consumer's cost of electricity and no more than 3% if nuclear power growth to at least 250 GWe is realized.

TABLE 1.7.7. Comparison of Levelized Waste-Management Unit Costs for the Program Alternatives at a 7% Discount Rate, ¢/kWh

Case	Nuclear Power Growth Assumption	Proposed Program (Geologic Disposal Starting 1990 - 2010)	Alternative Program (Disposal Starting 2010 - 2030)	No-Action Alternative
1	Present Inventory Only	0.2	0.2	0.08
2	Present Capacity and Normal Life	0.1	0.1	0.06
3	250 GWe system by Year 2000 and Normal Life	0.06 to 0.09	0.07	0.05
4	250 GWe System by Year 2000 and Steady State	0.07 to 0.08	0.07	NA[(a)]
5	500 GWe System by Year 2040	0.06 to 0.08	0.07	NA

(a) NA = not applicable.

1.8 CONCLUSIONS

Based on the environmental impacts evaluated in this Statement, it is concluded that a decision to proceed with the proposed action, that is, development of a programmatic strategy favoring the disposal of commercially generated radioactive wastes in deep geologic repositories, is warranted. This conclusion applies whether the wastes are generated in the once-through or in the reprocessing fuel cycle option.

This conclusion is based on the information contained within this document (and appropriate references) which indicate that the environmental impacts of the program alternatives are similar. The consequences of delaying implementation of a specific disposal technology should not result in any appreciable change in the near-term environmental effects. The decision to emphasize mined geologic repositories as the primary disposal technology is similarly based on an evaluation of the long term effects which indicates that mined geologic disposal and those technologies which justify further consideration would have relatively equal environmental impact. It is recognized that although the level of knowledge of the alternative technologies is not comparable, sufficient evidence exists to support that there is little likelihood that these technologies would be superior, from an environmental perspective, to the geologic alternative.

The no-action alternative is undesirable because of the temporary nature of present storage of wastes, the need to construct additional facilities for extended storage as present facilities reach their design lifetime, and because the no-action alternative is contrary to the presidential proclamation and could be construed as contrary to the mandate given DOE by law. Analysis of the no-action alternative in this Statement has not considered possible failures that could occcur if present facilities designed for temporary use were to be used indefinitely. It is possible that no-action could result in unacceptable safety and environmental consequences.

More specifically, regarding the three program alternatives considered in the Statement, the following conclusions can be drawn:

- Radiation dose accumulations increase as the size of the nuclear system increases. Neither the dose accumulation nor the health effects are significantly different for the program alternatives in either the once-through or reprocessing cycles. The dose accumulation with reprocessing is much larger (principally because of doses from radioactive material in dissolver off gas that is released to the environment)[a] than with the once-through cycle. For comparison, this amounts to 0.5% of the regional and 0.003% of the worldwide dose from natural causes over the same period in the highest nuclear growth case examined here.

(a) Estimated dissolver off gas releases are within the EPA Standard for ^{85}Kr and ^{129}I which becomes effective in 1983 (40CFR190.10).

- Resource commitments also increase with increasing size of the nuclear system. With the once-through cycle, resource requirements for the alternative program range up to 2 to 3 times higher than for the proposed program. With the reprocessing cycle, resource requirements for the alternative program are about the same to slightly higher than for the proposed program. For all cases, resource requirements are a small fraction of current U.S. production rates.
- Waste management costs increase as the size of the nuclear system increases, the waste management cost range is significantly higher for the alternative program than for the proposed program. With the reprocessing cycle, the cost ranges are about the same for both alternatives. The no-action alternative costs fall in the low end of the cost range for the proposed program with the once-through cycle. When costs are compared on the basis of levelized unit costs at a 7% discount rate, differences between the alternative and proposed programs and differences between reprocessing and the once-through cycle are slight.
- Societal risk from several events with low probability and high consequence in the long term following geologic repository closure was determined to be small in comparison to other societal risks even if large errors in judgement of the probability of occurrence were made. This conclusion appears valid even if no credit is taken for effects of multiple engineered and geologic barriers that will be employed to further assure containment and isolation.

With respect to the alternative waste disposal technologies considered in this Statement, the following conclusions can be drawn:

- A mined geologic repository is the preferred alternative based on evaluation of radiological effects during the operational period, non-radiological effects on the human environment, status of development, conformance with existing National and international law, independence from future development of the nuclear industry and potential for corrective or mitigating actions. The potential for and consequences of unplanned events in the long term require further investigation. The only category in which an alternative technology might offer an advantage would be the radiological effects during the post-operational period for which space disposal appeared more preferable. However, this long term advantage would be more than offset by near term disadvantages.
- Subseabed disposal appears promising enough to warrant further detailed examination. The potential for and consequences of unplanned events in the long term also require further investigation for this option. Studies of the anticipated environmental

(a) This disposal technology would not be capable of accommodating the full range of waste types. An alternative technology, i.e., geologic disposal, would be required for large quantities of solid waste. Thus, this alternative should be viewed as complementary to geologic disposal.

effects associated with special port facilities and tranportation links will be made. The practicality of pursuing this concept, recognizing current National and international laws and agreements will be further analyzed.

- Very deep hole disposal warrants some additional study as a possible backup for HLW disposal only. Further development should emphasize the ability for corrective or mitigating actions available.[a]
- Space disposal may be profitably studied for its application to special disposal concerns, e.g., more remote isolation of long lived and environmentally mobile radionuclides such as ^{99}Tc and ^{129}I.[a] However, the overall impact on the total waste management system will need to be carefully evaluated to determine if such separation would provide overall benefit.
- Other technologies studied (island, mined repository, transmutation, rock melt, ice sheet and well injection) either have no clear advantage over geologic disposal, or provide no additional complementary function and, in some cases, are clearly less desirable.

It can be argued that a delay in the program strategy, which would allow for a longer period of R&D, could conceivably reduce the probability of failure of the chosen disposal system by producing more knowledge and a greater diversity of choice in selecting a disposal method. DOE concludes that the likelihood of this occurring is small. In addition, the DOE program allows for a continuing broad based R&D effort, the investigation of a broad range of alternative media, and technical conservatism in program implementation.

Because this Statement is not site-specific it will be necessary to make other environmental analyses addressing the possibility of adverse impacts associated with specific sites and facilities at such time as the program reaches such decision points.

Recovery of the full costs of research and development and implementation of waste management and disposal for all modes of operation considered in this EIS, with the assumption of continued nuclear power growth to 250 GWe, resulted in a 2 to 3% increase in estimated average cost of electrical energy to the consumer. (Complete cessation of nuclear power generation at the end of 1980 would result in a significantly higher cost of waste management per unit of power produced.)

In summary, there appear to be no environmental issues that would reasonably preclude pursuit of a program strategy favoring disposal of commerically generated radioactive wastes in deep geologic repositories (regardless of nuclear power growth assumptions). Thus the proposed action of conducting R&D leading to disposal of radioactive wastes in deep geologic repositories is believed to be fully supported.

(a) This disposal technology would not be capable of accommodating the full range of waste types. An alternative technology, i.e., geologic disposal, would be required for large quantities of solid waste. Thus, this alternative sould be viewed as complementary to geologic disposal.

REFERENCES FOR CHAPTER 1

Code of Federal Regulations. Title 40, Part 190.

U.S. Department of Energy. 1980. Statement of Position of the United States Department of Energy in the Matter of Proposed Rulemaking on the Storage and Disposal of Nuclear Waste. DOE/NE-0007, Washington, D.C.

CHAPTER 2

INTRODUCTION

The United States Department of Energy (DOE) has the responsibility to develop technologies for management and disposal of certain classes of commercially generated radioactive wastes (namely high-level and transuranic). To provide input to the decision on a planning strategy for disposal of these radioactive wastes, this Statement presents an analysis of environmental impacts that could occur if various technologies for management and disposal of such wastes were to be developed and implemented.

In this Statement, which often has been referred to as a generic environmental impact statement (GEIS), the various options for permanent waste isolation are examined in a generic or general sense rather than in a site-specific sense. Various concepts are examined for the environmental impacts that their implementation might cause at any non-specific or generic locations. Upon selection of specific locations for waste disposal using the proposed approach, future site-specific environmental analyses will be prepared.

Section 2.1 describes the relationship of this environmental impact statement to other waste management decisions and associated environmental impact statements. This section also outlines the relationship of the President's recent message on disposal of radioactive wastes to the forthcoming National Plan for Nuclear Waste Management.

Section 2.2 describes the structure and content of this Statement. This section also describes the relationship of this Statement's format to those decisions that are to be made (for which this EIS will serve as the environmental input).

Section 2.3 discusses future decisions related to the disposal of commercial radioactive waste.

2.1 RELATIONSHIP TO OTHER WASTE MANAGEMENT DECISIONS

This Statement, Management of Commercially Generated Radioactive Waste, analyzes impacts of high-level and transuranic waste management following removal of spent light water reactor fuel[a] from nuclear power plants (reactors). The responsibility for developing technology for disposal of radioactive wastes has been assigned to the DOE by the U.S. Congress. The primary emphasis of this Statement is on the safe, permanent isolation of radioactive wastes. Also discussed are interim waste storage, treatment, transportation and facility decommissioning as they relate to a decision on the proposed method of waste disposal.

The basic waste management steps in the commercial LWR nuclear fuel cycle are shown in Figure 2.1.1. The heavy solid lines show waste streams covered in this Statement. Airborne

(a) All but one of the large commercial power reactors operating in the U.S. today are of the light water reactor (LWR) type.

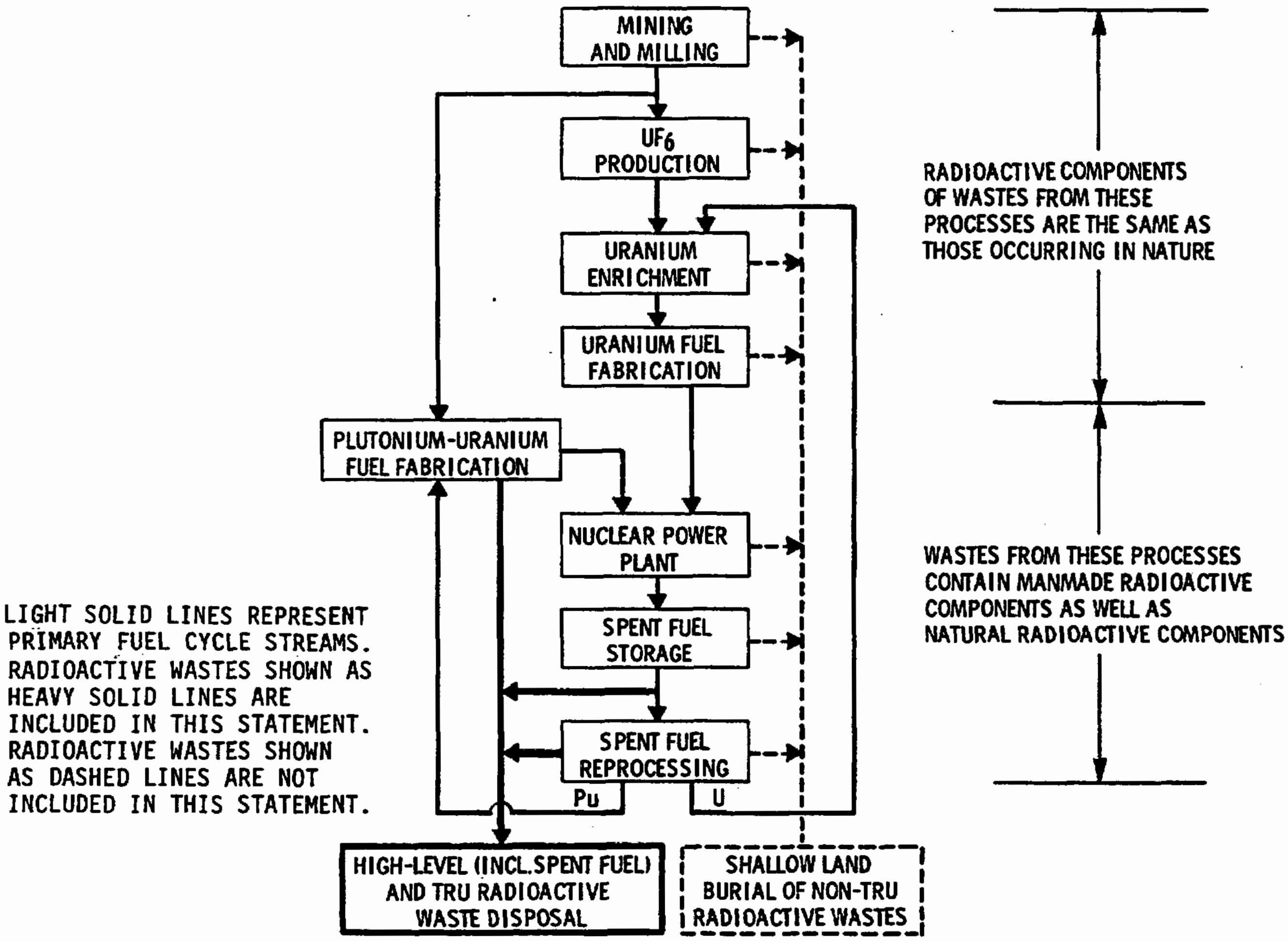

FIGURE 2.1.1. Processes and Waste Streams in the Commercial Fuel Cycle

wastes from spent fuel storage, reprocessing and plutonium-uranium fuel fabrication are also also covered. In addition to these wastes, a number of other radioactive wastes must be properly managed and disposed. This section describes the status of program and environmental statements covering these other wastes and also the status of statements covering broad areas (e.g., spent fuel storage and transportation) that are partially included in the overall system addressed in this Statement.

2.1.1 Mining and Milling

Mining and milling operations are currently regulated by either the Nuclear Regulatory Commission (NRC) or by Agreement States (states which have entered into an agreement with NRC pursuant to Section 274 of the Atomic Energy Act of 1954 as amended (42 U.S.C. 2021) under which the state government assumes regulatory authority and responsibility). Environmental impacts are considered programmatically in Uranium Milling, NUREG-0511 (NRC 1979a). Individual EISs have been prepared for each operation licensed. An example is Final Environmental Statement Related to the Plateau Resources Limited Shootering Canyon Uranium Project, NUREG-0583 (NRC 1979b).

2.1.2 Uranium Enrichment

To date, two impact statements have been prepared relative to uranium enrichment:

Final Environmental Statement, Expansion of U.S. Uranium Enrichment Capacity, ERDA-1543 (ERDA 1976)

Final Environmental Impact Statement, Portsmouth Gaseous Diffusion Plant Site, Piketon, Ohio, ERDA-1555 (ERDA 1977a).

2.1.3 Uranium Fuel Fabrication

No generic statement has been prepared for uranium fuel fabrication. This operation is covered by individual statements for specific facilities. Examples of such impact statements are:

Environmental Impact Appraisal, Westinghouse Nuclear Fuel Columbia Site Commercial Nuclear Fuel Fabrication Plant, Columbia, South Carolina, April 1977.

Environmental Impact Appraisal of Nuclear Fuel Services Erwin Plant, Erwin, Tennessee, January 1978.

2.1.4 Low-Level Waste

At present, low-level wastes are regulated by the NRC or by Agreement States. In the event legislation is passed giving DOE any responsibilities related to disposal of low-level wastes from commercial activities, a programmatic environmental statement would be prepared. Environmental impacts of low-level waste activities are described in various NRC documents such as Final Generic Environmental Statement on the Use of Recycled Plutonium in Mixed Oxide Fuel in Light-Water Cooled Reactors, NUREG-002 (NRC 1976).

2.1.5 Spent Fuel Storage

In October 1977, DOE announced a Spent Fuel Storage Policy for nuclear power reactors. Under this policy, U.S. utilities would be given the opportunity to deliver spent power reactor fuel to the U.S. Government in exchange for payment of a fee. The U.S. Government would also be prepared to accept a limited amount of spent fuel from foreign sources when such action would contribute to meeting U.S. nonproliferation goals. A bill was submitted to Congress to authorize action required to implement the Spent Fuel Storage Policy. This bill, known as the "Spent Nuclear Fuel Act of 1979," would authorize the Secretary of Energy to acquire or construct one or more away-from-reactor (AFR) storage facilities. The Secretary would be authorized to accept title to and provide interim storage and ultimate disposal for domestic spent fuel and limited amounts of foreign spent fuel. A final programmatic EIS, Final Environmental Impact Statement, U.S. Spent Fuel Policy, DOE/EIS-0015 (DOE 1980a) has been issued which addresses the environmental impacts of various options regarding the interim storage of domestic fuel, the receipt of some foreign fuel, and the fee methodology for determining the charge for spent fuel storage.

With regard to receipt and storage of foreign spent fuel, the impacts described in the present Statement cover a range of future domestic power production which is sufficiently broad that it would encompass any possible impact due to quantities of spent fuel which might be shipped from other countries to the U.S. Foreign spent fuel which could be returned to the United States for storage or possible disposal would be predominately the LWR type.

Because a decision has been made to implement the Spent Fuel Storage Policy if authorized by Congress, an AFR spent fuel storage facility EIS will be prepared to provide the environmental input into the selection of facilities to meet the demand for spent fuel storage.[a] The environmental effects associated with the acquisition, construction and/or operation of the facilities and the transportation effects associated with the available options would be evaluated in this environmental documentation.

2.1.6 Transportation

The NRC and the Department of Transportation (DOT) regulate the transportation of radioactive waste. Transportation and packaging criteria and standards are outlined in the Code of Federal Regulations (10 CFR 71 and 49 CFR 170-189). The environmental impacts of transportation activities are addressed in Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes, NUREG-0170 (NRC 1977).

The present Statement specifically examines the transportation of post-fission wastes (spent fuel, high-level waste and TRU waste) from commercial LWR fuel cycle facilities to both interim storage locations and final isolation sites.

2.1.7 Alternative Reactor Types

The present Statement discusses and compares the characteristics of the wastes generated in the management of thorium fuels from the Light Water Breeder (Conversion) Reactor and High-Temperature Gas Cooled Reactor fuel cycle with those obtained from the LWR fuel cycle. No decisions to construct such reactors would be made before consideration is given to the disposal of waste from these reactors. However, the impact of wastes which would be generated by a future Liquid Metal Fast Breeder Reactor (LMFBR) fuel cycle is not analyzed here. They were addressed in Final Environmental Statement, Liquid Metal Fast Breeder Reactor Program, ERDA-1535 (1975a).

2.1.8 Wastes From National Defense Activities

High-level waste from national defense activities is currently being stored on DOE reservations in Idaho, South Carolina, and Washington. EISs that consider the short term storage of these wastes at these sites have been prepared (ERDA 1975b, 1977b, and 1977c, respectively).

(a) The Notice of Intent regarding prepartion of the spent fuel storage facilty EIS was issued in the Federal Register on August 15, 1980 (45FR54399).

Since waste forms and conditions are different at the three sites, programmatic statements covering development programs for final waste treatment and final disposal are being prepared for each site.

Transuranic wastes resulting from national defense activities are also stored at the sites listed above and at Los Alamos, New Mexico; the Nevada Test Site; and the Oak Ridge National Laboratory in Tennessee. Statements covering waste treatment and final disposal of material now stored at these sites will also be prepared.

This Statement does not directly address management and disposal of radioactive wastes related to national defense programs. However, in a generic sense, systems that can adequately dispose of commercial radioactive wastes have the capability to adequately dispose of wastes resulting from defense programs.

2.1.9 National Plan for Nuclear Waste Management

The President, in his nuclear waste policy statement of February 12, 1980, stated that the safe disposal of radioactive waste, generated from both national defense and commercial activities, is a national responsibility. In fulfillment of his responsibility, the President has directed the Department of Energy, in its role as lead agency for the management and disposal of radioactive wastes, to prepare a comprehensive National Plan for Radioactive Waste Management. This National Plan is being prepared in cooperation with other involved Federal agencies, primarily the Departments of Interior and Transportation, the Environmental Protection Agency, and the Nuclear Regulatory Commission. The State Planning Council, which was established by the President, will also be involved in the development of the National Plan.[a] This Plan will provide a road map for all parties and give the public an opportunity to review DOE's entire program. The Plan will be comprehensive in scope and include relevant activities of the Federal agencies, states, and local governments. The Plan will cover all types and sources of radioactive waste and present the strategy and sequence of events to manage effectively and dispose of radioactive wastes and associated regulatory activities.

Methods of communication between and among Federal agencies, states and local governments, and the general public will be presented to show current and proposed interactions and the nature and degree of public participation in the planning and decisionmaking process, including the preparation of the National Plan. The National Plan will be updated every 2 years in recognition of and response to results of R&D programs, actual operations, and guidance from institutions such as Federal agencies, state governments, the State Planning Council and others that might be affected by programs and proposed actions.

A draft of the comprehensive National Plan will be distributed by the Secretary of Energy in the fall of 1980, for congressional and general public review and comment. After reviewing public comments and revising the National Plan, a final version of the National

(a) The Council will provide advice and recommendations to the President and the Secretary of Energy on nuclear waste management issues.

Plan, including a summary of the public comments, will be issued in 1981. The National Plan will be used by the Congress, Federal agencies, and the general public to understand the scope, direction, and interrelationship of activities and the progress being made to implement the President's policy.

2.2 STRUCTURE AND CONTENT OF STATEMENT

This Statement describes the character and quantities of the wastes to be managed from various nuclear power generation scenarios and identifies the environmental impacts (i.e., radiological effects, non-radiological effects, resource requirements, socioeconomic impacts, costs, institutional issues) associated with the management of these wastes. The power generation scenarios considered and the scope of the analysis are detailed in Section 3.2. As DOE has the responsibility for selecting a programmatic strategy for the management of commercial radioactive wastes, this Statement presents an analysis of alternative waste management programs for meeting this requirement. The three programmatic strategies presented in the Statement are:

- Proposed Action. The research and development program for waste management will emphasize use of mined repositories in geologic formations in the continental U.S. capable of accepting radioactive wastes from either the once-through or reprocessing cycles (while continuing to examine subseabed and very deep hole disposal as potential backup technologies). This action will be carried forward to identify specific locations for the construction of mined repositories. The proposed action does not preclude further study of other disposal techniques. For example, the selective use of space disposal for specific isotopes might be considered.

- Alternative Action. The research and development program would emphasize the parallel development of several disposal technologies. This action implies an R&D program to bring the knowledge regarding two or three disposal concepts and their development status to an approximately equal level. Based upon the Department of Energy's current evaluation, the likely candidate technologies for this parallel development strategy would be:
  1) geologic disposal using conventional mining techniques
  2) placement in sediment beneath the deep ocean (subseabed)
  3) disposal in very deep holes.

  At some later point, a preferred technology would be selected for construction of facilities for radioactive waste disposal.

- No Action Alternative. This alternative would eliminate or significantly reduce the Department of Energy's research and development programs for radioactive waste disposal. Under this alternative, existing spent fuel would be left indefinitely where it is currently stored and any additional spent fuel discharged from future operation of commercial nuclear power plants would likewise be stored indefinitely in water basin facilities either at the reactors or at independent sites.

Beyond the selection of a program strategy, DOE must determine the pace and manner in which to pursue the selected program. To this end, this Statement examines 1) a range of dates for the availability of a mined geologic repository and 2) a variety of candidate repository media (salt, basalt, granite, shale).

The main body of the text (Volume 1) is divided into eight chapters. Chapter 3 presents the program alternatives under consideration and outlines the technological and environmental bases for the analysis. Discussions of natural background radiation and the concept of risk are included to give the reader additional perspectives from which to view the material in the Statement. Non-technical concerns relevant to waste management are also identified for the purpose of airing such issues, which will have to be addressed in any ongoing plan.

Chapter 4 describes the wastes and analyzes the various activities required prior to final disposal on a unit basis (e.g., per GWe-yr, per Kg HM, per facility). The processes of waste treatment, storage, transportation and facility decommissioning are addressed and their impacts are presented. Chemical resynthesis and partitioning, items included in the draft in the presentation of disposal techniques, now appear in the discussion of waste treatment alternatives. A discussion of the relationship between predisposal activities and the individual disposal technologies is also included in Chapter 4.

Chapters 5 and 6 examine the mined geologic disposal concept and alternative disposal technologies, respectively. For consistency of presentation, discussion of each disposal concept addresses the same topic areas:

- Concept and System Description
- Status of Technical Development and R&D Needs
- Disposal Facility Description
- Environmental Impacts of Construction and Operation
- Environmental Impacts Over the Long Term
- Cost Analysis
- Safeguard Requirements.

The depth of the presentation, however, is not identical for the various disposal alternatives for two reasons. First, the extent to which a disposal concept can be examined is a function of the degree to which the concept has been researched, developed, and reported in previous studies. Accordingly, mined geologic disposal is more fully described than the other disposal modes. Secondly, an assessment of the impacts from implementing a disposal alternative is predicated on having data that can be substantiated. The existing data base for mined geologic disposal is significantly more extensive than for the other concepts; hence, a more detailed analysis of impacts is possible.

At the end of Chapter 6, a comparison is made of the nine disposal technologies presented in Chapters 5 and 6 on the basis of several environmental and policy-related criteria.

Chapter 7 outlines the trade-offs between the program alternatives (identified in Chapter 3), with emphasis on the entire waste management system. The points of comparison of the alternative actions deal with nuclear power growth assumptions, fuel cycles, waste volumes, and environmental impacts based on the material in Chapters 4, 5, and 6.

Chapter 8 is a glossary of key environmental, geologic, and waste technology-related terms and acronyms.

Volume 2 is a compilation of appendix material. Volume 3 is a presentation of written public and agency comments and Hearing Board recommendations on the draft Statement and responses to these comments and recommendations.

During the reviews of the draft Statement, some commenters urged that the option of shutting down all nuclear power plants be considered in the final Statement. Although such an action is beyond the authority of the DOE and can be considered only by the NRC or by the U.S. Congress, this Statement does present an analysis of managing only present inventories of spent fuel. While the availability of adequate waste management methods should be considered by these institutions in contemplating such an action, many other far-broader issues, such as national energy and economic requirements and the overall safety and environmental impacts of other energy systems, would also need to be considered. Due to the extent of DOE's authority, the scope of this environmental impact statement is limited to consideration of the impacts of successfully implemented programs for research and development leading to permanent disposal of present and future high-level and TRU radioactive wastes.

2.3 OTHER DECISIONS CONCERNING DISPOSAL OF COMMERCIAL WASTES[a]

The decisions that the DOE now faces and for which the analysis in this Statement will provide environmental input will not automatically lead to the placement of radioactive wastes in any specific location. As the program of research and development and examination of specific candidate locations proceeds, further decisions will be required relative to potential environmental impacts.

The National Environmental Policy Act of 1969 (NEPA 1969), as implemented by the regulations of the Council on Environmental Quality (CEQ 1978) and the DOE guidelines (DOE 1980b), requires that environmental consequences be considered in Department planning and decisionmaking. In adopting a strategy for disposal of high-level radioactive wastes, the DOE will undertake actions having potential environmental consequences. The potential environmental effects of these actions and their significance vary. Actions range from the decision adopting the overall strategy for waste disposal (involving a major resource commitment which ultimately may have a spectrum of potential environmental effects specific to that strategy) to the selection of specific sites and facilities for waste disposal purposes. Other actions include the conduct of research (data gathering and analysis) which may have little environmental effect but which may have important technological, cost, and time implications on long-term waste disposal.

Using the CEQ regulations and the DOE guidelines, a NEPA implementation plan, which is integrated with overall DOE planning and decisionmaking, has been developed for the deep mined geologic disposal strategy. Figure 2.3.1 graphically demonstrates the various steps associated with integration of the NEPA plan and the overall decisionmaking process.

The DOE's NEPA implementation plan is based on the "tiered" approach, which is designed to eliminate repetitive discussions of the same issues and to focus on the actual issues ripe for decision at each level of environmental review. This approach allows coverage of general matters in broad environmental impact statements (EISs) with subsequent narrower EISs or environmental assessments (EAs) incorporating by reference the general discussions and concentrating solely on the issues specific to the subsequent decision.

The NEPA implementation plan identifies the major decision points in the program to assure that appropriate environmental documentation is completed prior to each such decision and prior to the conduct of activities that may cause an adverse environmental impact or limit the choice of reasonable alternatives. The first major decision process is selection of a program strategy for disposal of nuclear waste. This Statement serves as the NEPA input for this first decision.

(a) Much of the material in this section was taken from the recent DOE Statement of Position in the NRC rulemaking proceedings on nuclear waste storage and disposal (DOE 1980c). The Statement of Position described in DOE's proposed research and development program and was prepared pursuant to the initiation of the rulemaking proceedings. The present Statement, upon issuance as a final impact statement, will become part of the record of the rulemaking proceedings.

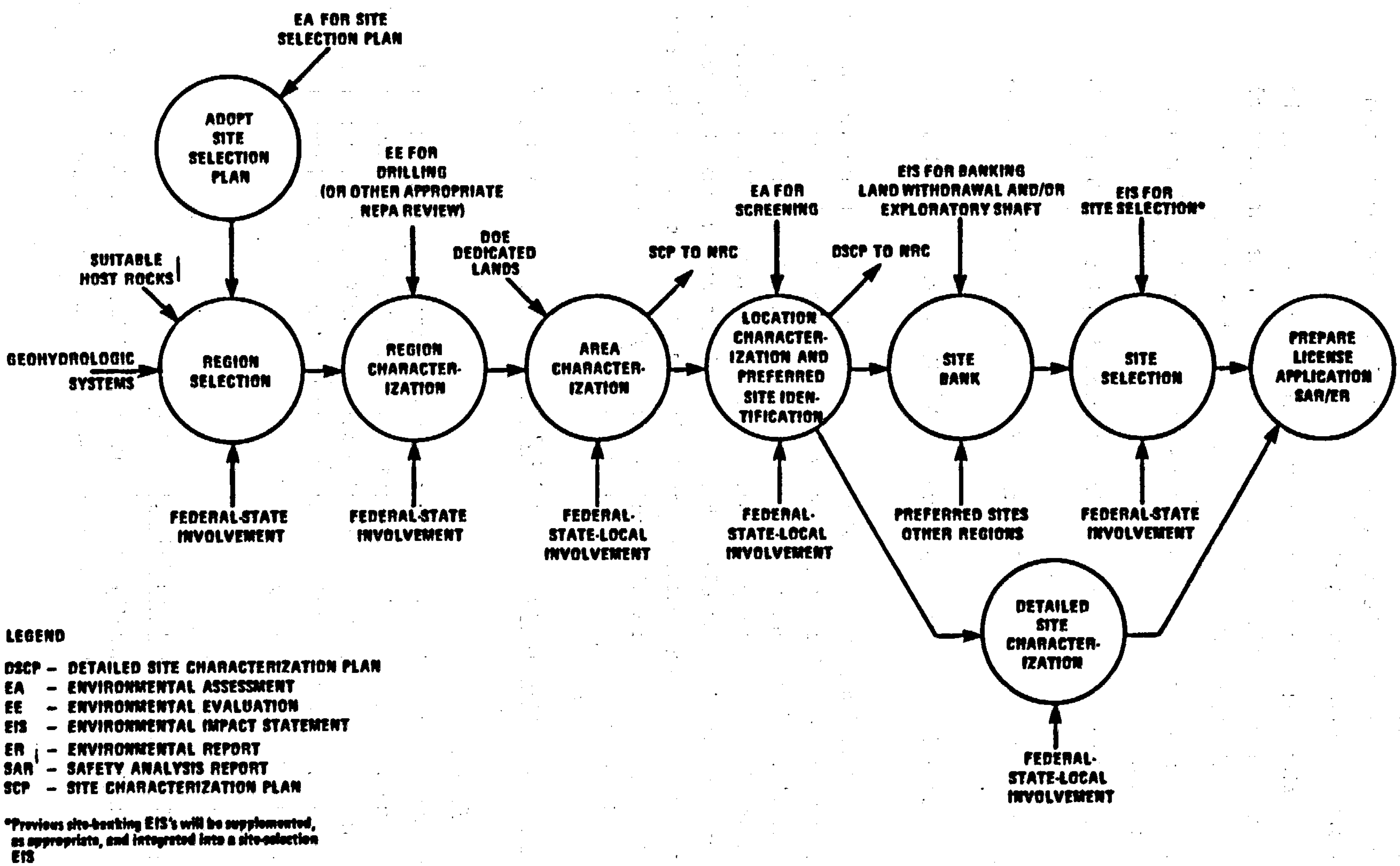

FIGURE 2.3.1. Site Characterization and Selection Process

The second major decision process is that involving the selection of sites for the disposal of nuclear waste assuming the mined geologic option. The major decision points in such a site-selection process are:

1. Adoption of a National Site Selection and Characterization Plan including the national screening for potential regions and selection of areas (approximately 2,590 square kilometers, or 1,000 square miles) for further study.
2. Identification of locations (26 to 78 square kilometers, or 10 to 30 square miles) for in-depth study.
3. Selection of a preferred site(s) for banking,[(a)] including the possible development of an early shaft.
4. Acquiring an interest in land sufficient to protect potential sites from other uses.
5. Selection of a candidate site to propose to NRC for licensing as the first repository.

At each of these decision points, the DOE will consider the appropriate NEPA documentation. While the appropriate NEPA documentation is being prepared for the various decision points, program activities, including site characterization activities, that have been analyzed in previous NEPA documents may continue. In addition, further site characterization activities may continue if it is clear, based on the DOE's review, that they do not 1) have significant adverse environmental impact or 2) limit the choice of reasonable alternatives (40 CFR 1506.1). These activities could include environmental studies, routine geophysical studies, shallow drilling, and borehole drilling.

2.3.1 The DOE's National Environmental Policy Act Implementation Plan[(b)]

2.3.1.1 Program Strategy

The environmental effects of implementing a program strategy are addressed in this final EIS on Management of Commercially Generated Radioactive Waste. Based upon the analyses of nine disposal concepts, mined geologic disposal is identified as the preferred technical alternative and the proposed action is the selection of a program strategy emphasizing geologic disposal in a mined repository.

2.3.1.2 Site Selection Process

National Site Characterization and Selection Plan

The DOE proposes to adopt formally the current National Waste Terminal Storage Site Characterization and Selection Plan as the comprehensive National Site Characterization and

(a) Protecting a potential repository site(s) from conflicting uses until such time as a final site(s) is selected.
(b) Section 5.2 and Appendix B.7 discuss the technical considerations of repository site selection.

Selection Plan. The current plan, described elsewhere (DOE 1980c), will be followed pending adoption of a formal plan. An EA is being prepared as input to the decision on whether to adopt or modify this plan.

The proposed plan includes:

- The methodology for identifying geographic regions for site studies.
- The methodology and criteria for screening these regions for areas, locations, and candidate sites to be studied in detail.

The environmental impacts of the methodology and criteria in the proposed plan and their reasonable alternatives will be assessed. In addition, the selection of areas for further study and the anticipated range of site characterization activities, including the environmental impacts of typical surface and subsurface activities in several environmental settings, will be analyzed. Similarly, the criteria proposed to be used to qualify and disqualify sites will be discussed.

It is believed that an EA, and not an EIS, is the appropriate level of NEPA review, since it is unclear that the decision will result in significant environmental impacts. However, upon completion of the EA, a decision will be made regarding the need to prepare an EIS. The Department of Energy will consider the results of the NEPA review prior to deciding whether to adopt or modify the proposed plan. The adopted site characterization process will be repeated in diverse geologic environments and different host media until four to five sites have been qualified.

Identification of Locations

Following completion of area studies for a particular region, in accordance with the National Plan, an EA will be prepared as input for a decision to narrow the investigations to a limited number of locations. The site-selection process to date will be described, and the environmental factors pertinent to the proposal to limit more comprehensive exploratory activities to the preferred locations will be analyzed. A comparison of environmental factors for preferred and alternate locations, based on data commensurate with the level of site-specific information available, will be provided and the environmental impacts of the range of potential exploratory activities anticipated in the location studies will be considered.

Here, too, it is believed that an EA is the appropriate level of NEPA review, since it is unclear that this decision will have environmental significance. Upon completion of the EA, a decision will be made regarding the need to prepare an EIS.

Identifying Preferred Sites for Banking/Early Shaft

At the conclusion of the location studies, the DOE will propose one or more of the sites in a location as a preferred site to be banked. Because a banked site ultimately may become the location of a repository, it is appropriate to prepare an EIS prior to the decision to bank the preferred site(s). This EIS also would provide input to a decision to acquire an interest in the site(s), if necessary, in order to maintain the integrity of the site through the site-selection process.

Using a general conceptual design for the appropriate media (a site-specific design will not be developed until after the candidate site is selected), the EIS will evaluate the potential environmental impacts of 1) a conceptual repository at the alternate sites within the region and 2) the detailed site characterization activities which may be required at each of the alternate site(s), including the possible construction of an early shaft, if required.

Although the general conceptual design will not be site-specific, it will be in an advanced stage of development relative to the medium in which the potential candidate sites are located. This will allow adequate analysis of the potential environmental impacts associated with a conceptual repository at each of the alternative sites. In addition, the interaction of waste package options with the geologic medium will be assessed in each site-banking EIS.

Site Selection

Following the banking of sites in several media, a site will be selected for a license application for the first repository. The EISs previously prepared for site banking will be supplemented, as appropriate, in an integrated EIS, which will provide a comparative environmental analysis of the alternative sites. This EIS will incorporate by reference the site-banking EISs and include any significant new information obtained since the preparation of the earlier EISs. The site-selection EIS also will serve as input to the environmental report submitted to NRC with the license application.

2.3.1.3 Land Acquisition

After a site-selection decision, the DOE may take steps to permanently acquire the site. The site banking EISs, as supplemented in the site-selection EIS, will be used as input to the land acquisition decision.

REFERENCES FOR CHAPTER 2

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CHAPTER 3

DESCRIPTION OF PROGRAM ALTERNATIVES AND BACKGROUND

This section describes the major action proposed by the Department of Energy for which this environmental impact statement was prepared, namely the selection of a programmatic strategy emphasizing geologic disposal in a mined repository as the technology for disposal of high-level radioactive wastes. Two programmatic alternatives to this proposed action are also described. In addition, this section provides the reader with a description of the technical and environmental bases for the analyses which follow in succeeding sections. Since radiation exposure is a central concern in the management and disposal of nuclear wastes, background information about radiation and the approaches used to assess radiological risk are presented. Finally, "non technical" issues are discussed to inform the reader about the broad social, political, and institutional concerns which cut across specific technical concerns about nuclear waste.

3.1 PROPOSED ACTION AND PROGRAM ALTERNATIVES

As part of its responsibility for developing the technology required for managing certain classes of radioactive wastes, the Department of Energy proposes to take a major agency action: selecting an appropriate programmatic strategy leading to the disposal of commercial radioactive waste in a fashion that provides reasonable assurance of safe, permanent isolation of these materials.

This major action involves two specific components at this time. The first is the selection of geologic disposal in a mined repository as the technology for emphasis in a research and development program from among the various concepts that have been proposed. The second decision concerns the nature and extent of the research and development program to be undertaken, given the designation of geologic disposal as the technology for emphasis.

In considering alternative methods that might be employed for permanent isolation of radioactive materials, this EIS identifies and examines nine disposal technologies. These technologies, fully characterized in Chapters 5 and 6, are:

1) geologic disposal using conventional mining techniques
2) disposal in very deep holes
3) disposal in a mined cavity that results in rock melting
4) disposal in repositories located on an island
5) disposal in sediments beneath the deep ocean in the subseabed
6) disposal in an ice sheet in the Arctic or Antarctic
7) disposal in an injection well
8) disposal by partitioning of reprocessed waste and transmutation of actinides
9) disposal by projection into outer space.

In considering the nine disposal technology concepts, a variety of nuclear wastes is considered. Each concept needs to be evaluated in terms of capability to handle both spent fuel (as a waste) and waste from fuel reprocessing. Further, the ability of these technologies to accommodate transuranic (TRU) wastes is evaluated (see Section 6.2). As shown in Table 3.1.1, not all of the technologies are capable of handling all three categories of waste efficiently. Nonetheless, some of these technologies may be useful for special purposes such as the disposal of very long-lived radioactive substances. Some concepts are rated impractical because of special handling requirements, anticipated cost, environmental risks and current capabilities to implement the technology.

TABLE 3.1.1 Potential Ability of Technology to Handle Waste Type

Technology	Unprocessed Spent Fuel	High-Level Reprocessing Waste	TRU Waste
Geologic	Yes	Yes	Yes
Very Deep Holes	Yes	Yes	I
Rock Melting	No	Yes	No
Island	Yes	Yes	Yes
Subseabed	Yes	Yes	I
Ice Sheet	Yes	Yes	I
Injection Well	No	Yes	No
Transmutation	No	Yes	No
Space	I	Yes	I

LEGEND: Yes--Concept applies
No--Concept will not work
I--Concept impractical.

Evaluation of these various technical alternatives for waste isolation has resulted in a finding that geologic disposal (placement of radioactive wastes in geologic formations using conventional mining techniques) is the preferred technology for research and development. However, the evaluation of these alternatives has led to the conclusion that two other disposal concepts deserve further examination as potential backup or ancillary technologies to geologic disposal: subseabed disposal (placement of wastes in sediments beneath the deep oceans), and very deep hole disposal (placement of wastes into very deep drill holes).

This Statement examines the ultimate environmental impacts of the Department of Energy's proposed action, a research, development and demonstration program emphasizing mined geologic repositories, as well as two alternative courses of action: 1) parallel development of several technologies to an approximately equal level prior to a decision on implementation and 2) the alternative of no action.

The Interagency Review Group (IRG) on Nuclear Waste Management in its report of March 1979, identified a number of alternative technical strategies, the environmental impacts of which are encompassed in the analyses contained in this Statement. The IRG Report recommended after considerable study and public input that:

- The approach to permanent disposal of nuclear waste should proceed in a stepwise basis in a technically conservative manner.
- Near-term program activities should be predicated on the tentative assumption that the first disposal facilities will be mined repositories, though nearer-term alternative approaches--subseabed and very deep hole disposal--should be given funding support.
- A number of potential sites in a variety of geologic environments should be identified, and action taken to reserve the option to use them if needed. Within technical constraints, actions should be taken to have several repositories operational before the end of the century in different regions of the country.

Beyond these recommendations, the IRG defined four alternative strategies for the development of repositories:

1. Strategy I provides that only mined repositories be considered for the first several repositories and that only geological environments with salt as the emplacement media would be considered for the first several repositories. As a result of past programs, a large body of information about salt as an emplacement medium exists. Thus, salt would be a probable choice for these repositories, since the speed of implementation of this strategy would likely rule out other media.
2. Strategy II is similar to the first, except that a choice of site for the first repository would be made from among whatever types of environments have been adequately characterized at the time of choice. However the first choice would still likely be from environments based on salt geology.
3. Strategy III provides that, for the first facility only mined repositories would be considered. However, three to five geological environments possessing a wide variety of emplacement media would be examined before a selection was made. Other technological options would be contenders as soon as they had been shown to be technologically sound and economically feasible.
4. Strategy IV provides that the choice of technical option and, if appropriate, geological environment be made only after information about a number of environments and other technical options has been obtained.

These strategies are associated with different amounts of time needed to achieve an operational repository, with Strategy I requiring the least amount of time and Strategy IV requiring the most time.

DOE, on the basis of the input from many sources, has formulated a proposed research, development and construction program for mined geologic repositories that incorporates the recommendations of the IRG Report. Environmental impacts that would be associated with each of these differing strategies and with differences in timing of implementation (i.e., immediate versus delay) are well within the envelope of the analyses reported in this Statement. Environmental consequences associated with Strategies I through III are bounded by the environmental analyses of the Proposed Action, while those associated with Stratety IV are within the envelope of analyses performed for the Parallel Development Alternative Action. This latter action also envelopes the environmental consequences associated with a "delayed action" strategy, i.e., delaying siting of a repository until enough is known about several technical alternatives. These analyses examine the environmental consequences of constructing, operating and decomissioning waste management facilities.

3.1.1 Proposed Action

The proposed research and development program for waste management will emphasize use of mined repositories in geologic formations capable of accepting radioactive wastes from either the once-through or reprocessing cycles. This program will be carried forward to identify specific locations for the construction of mined repositories. The rationale for the selection of mined repositories as the preferred concept is presented in Section 6.2.5.

Initially, site characterization programs will be conducted to identify qualified sites in a variety of potential host rock and geohydrologic settings. As qualified sites are identified by the R&D program, actions will be taken to reserve the option to use the sites, if necessary, at an appropriate time in the future. Supporting this site characterization and qualification program will be research and development efforts to produce techniques and equipment to support the placement of wastes in mined geologic repositories.

The Department of Energy proposes that the development of geologic repositories will proceed in a careful step-by-step fashion. Experience and information gained in each phase of the development program will be reviewed and evaluated to determine if there is sufficient knowledge to proceed to the next stage of development and research. The Department plans to proceed on a technically conservative basis allowing for ready retrievability of the emplaced waste for some initial period of time.

The proposed timing for emplacement of waste into geologic repositories calls for at least two operational facilities before the end of the century. This schedule reflects the need to expand the technical evaluation of a broader set of geologic media and multiple sites and to consider a possible regional approach to repository siting. Changes in timing for emplacement of wastes in geologic repositories because of environmental or other considerations is considered within the scope of the proposed action presented in this Statement.

Some support would be provided to further evaluate the alternatives of placement in deep ocean sediments and in very deep holes. The purpose of this support is to permit continued evaluation of these technology options as alternatives to geologic disposal. These options are considered as backups or complements to geologic disposal and are presently not planned for full development.

3.1.2 Alternative Action--Parallel Development

As an alternative to emphasis on geologic disposal, the research and development program would emphasize the parallel development of several disposal technologies. This action implies an R&D program to bring the knowledge regarding two or three disposal concepts and their development status to an approximately equal level. At some later point, a preferred technology would be selected for construction of facilities for radioactive waste disposal.

Based upon the Department of Energy's evaluation, the likely candidate technologies for this parallel development strategy would be:

1) geologic disposal using conventional mining techniques
2) placement in sediment beneath the deep ocean (subseabed)
3) disposal in very deep holes.

In order to develop several technologies in parallel, the range of approaches within each disposal technology would likely be narrowed to a single candidate approach.

The geologic disposal program would concentrate on a most preferred geohydrological system and, possibly, host rock. By narrowing the focus of the program, resources of time, money, and manpower would be made available to pursue the parallel development programs of the other two technologies.

In a similar fashion, the subseabed program would focus on a preferred system for waste emplacement and on a few locations.

The program activities for very deep hole disposal would eventually be focused on specific deep geohydrological systems and in specific regions of the country. Since adequate information about such deep systems is not currently available to do this, a program of study would need to be developed to acquire such information.

The strategy to develop several disposal technologies in parallel requires the use of extended term storage facilities since significant additional time would be required to bring the technologies of sub-seabed and very deep hole disposal to a level of development equivalent to that of geologic disposal. The main differences between the Proposed Action and the First Alternative Action are the degree of emphasis on geologic disposal and the timing of actual construction of waste disposal facilities.

3.1.3 No-Action Alternative

This alternative would eliminate or significantly reduce the Department of Energy's research and development programs for radioactive waste disposal. Under this alternative, existing spent fuel would be left indefinitely where it is currently stored and any additional spent fuel discharged from future operation of commercial nuclear power plants would likewise be stored indefinitely in water basin facilities either at the reactors or at independent sites. The Department of Energy does not consider this no-action alternative to be a reasonable course, since it offers no solution for the long-term period beyond the useful life of the water basins.

3.2 BASES FOR THE ANALYSIS

A number of bases for analysis must be established to assess environmental impacts associated with a nuclear waste disposal technology. This includes the identification and description of predisposal facilities necessary for waste management, as well as a description of the disposal facilities themselves. Further, the physical, biological and social environments into which these facilities will be placed must be characterized. However, total or net environmental impacts cannot be described completely by the effects of single facilities in the environment, so this Statement also analyzes complete waste management systems. The key assumptions associated with a systems analysis are those of nuclear power growth (i.e., amount of waste to be disposed) and the nuclear fuel cycles considered (i.e., kinds of waste to be disposed).

The general approach to environmental assessment used here investigates potential impacts associated with construction, operation (including potential accidents), and decommissioning of predisposal facilities (including treatment, transportation and storage of wastes) and the repository system itself. Physical protection requirements for safeguarding the wastes from theft or sabotage are also evaluated. Impacts resulting from nuclear waste disposal include those associated with resource commitments, ecological and atmospheric effects, radiological effects, socioeconomic effects, and the costs of waste management and disposal.

Predisposal facilities are discussed in Chapter 4, and geologic repositories are discussed in Chapter 5. Conceptual facilities are described, their impacts and costs of construction and operation are estimated, and safeguard requirements are evaluated. These conceptual facilities and impacts are described in detail in Technology for Commercial Radioactive Waste Management, DOE/ET-0028, April 1979 and Environmental Aspects of Commercial Radioactive Waste Management, DOE/ET-0029, April 1979. Summary descriptions and key results are presented in Chapters 4 and 5.

A description of the physical environments for the different facilities is given in Chapter 5 for geologic disposal and in Chapter 6 for alternative technologies. The biological and social environments used hypothetical or reference conditions which were assumed common to all geologic repositories and associated waste management facilities. For assessing general environmental and health effects for these facilities, a single reference environment was developed and is described in Appendix F. This reference environment provides the necessary description of environmental characteristics (e.g., demography, atmospheric dispersion patterns, surface waters, plant and animal communities) that serve as a baseline for generically estimating environmental impacts of waste management and disposal. Three reference environments were used to assess the socioeconomic impacts of the influx of workers associated with geologic repositories and related facilities, because socioeconomic impacts are particularly sensitive to variation in demography (Appendix G). The use of reference environments should not be construed as an endorsement of particular regions for siting waste management and disposal facilities but rather as convenient and realistic assessment tools. Different reference environments and bases for analyses were used in the case of alternative disposal technologies and are described where used in Section 6.1.

In Chapter 6, alternatives to geological disposal in mined continental repositories are described, evaluated, and compared.

In Chapter 7, the requirements and impacts for entire waste management systems for several different nuclear industry growth assumptions are described. These requirement and impact descriptions incorporate information about the individual waste management components (described in Chapters 4 and 5) into system simulation calculations.

The assumptions used regarding nuclear fuel cycles and industry growth as well as the basis for assessing resource commitments, ecological and atmospheric effects, radiological effects, socioeconomic impacts, potential accidents, physical protection, and costs of management and disposal of nuclear wastes are described in the following subsections.

3.2.1 Nuclear Fuel Cycle Assumptions

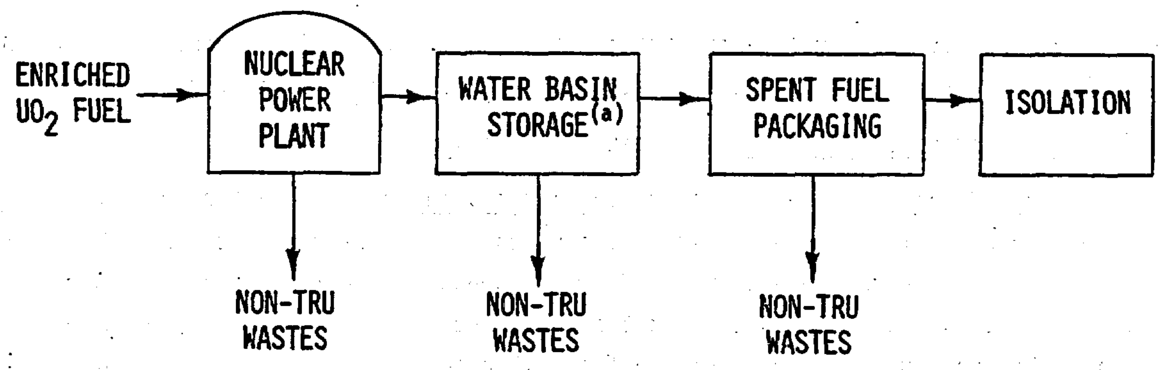
The waste management impacts of two basic light water reactor (LWR) fuel cycles are analyzed in this Statement. These are 1) the once-through fuel cycle where spent fuel is sent to disposal without reprocessing for recovery of residual energy potential, and 2) the reprocessing fuel cycle where spent fuel is determined to be a resource and is processed for recovery and use of the contained uranium and plutonium. A uranium-only recycle case (with plutonium remaining in the high-level waste or recovered and stored elsewhere) was considered in the draft of this Statement. However, because of the low likelihood that this fuel cycle would ever be implemented and because of comments to this effect received on the draft Statement, it has been deleted from this final Statement. Information on this fuel cycle may be found in DOE/ET-0028 and DOE/ET-0029.

3.2.1.1 Once-Through Fuel Cycle

A simplified diagram presenting the once-through cycle is shown in Figure 3.2.1. Spent fuel is stored until a qualified Federal waste isolation facility is in operation. Storage can occur either at the reactor site or at an offsite away-from-reactor (AFR) storage facility, also sometimes referred to as an independent spent fuel storage facility (ISFSF). Storage at an AFR is necessary if sufficient storage capacity is not available at nuclear power plant sites. At the AFR, only nontransuranic and gaseous wastes are generated[a] while the spent fuel is handled and stored. Thus, the only waste of concern to this Statement is the spent fuel itself. The following assumptions are made about the once-through fuel cycle.

- Although storage capacity in the nuclear power plant (reactor) basins will vary considerably and may be increased significantly for new plants, a given reactor basin will have, on the average, the capacity for seven annual discharges in addition to full core reserve. This capacity assumption results in away-from-reactor

(a) Strictly speaking, the radioactivity content in the wastes is "generated" during irradiation of the fuel in the nuclear power plant.



[a] WATER BASIN STORAGE IN EITHER REACTOR BASINS OR AFR FACILITIES

FIGURE 3.2.1. Once-Through Cycle

storage requirements that approximate the maximum requirements shown in a recent study when currently licensed expansion plans are all assumed to be implemented and full core reserve capacity is maintained (DOE/NE-0002 1980). Implications of variations in reactor storage capacity are discussed in the Final Environmental Impact Statement on U.S. Spent Fuel Policy (DOE/ET-0015 1980).

- To permit the spent fuel to cool down prior to dry encapsulation and disposal the spent fuel is stored for a minimum of 5 years in the nuclear power plant storage basins for the reference once-through fuel cycle. If a disposal facility is not available, the spent fuel remains stored at the reactor until the 7-yr capacity is filled, after which excess fuel older than 5 years is shipped (Section 4.5) to an AFR (Section 4.4) where it remains until a disposal facility is available.
- Spent fuel encapsulation (or packaging) facilities (Section 4.3) are located on the same site as the disposal facility. An alternative of encapsulating the spent fuel at the AFR and storing packaged spent fuel is also described in the predisposal system discussions in Sections 4.3 and 4.4.
- For purposes of estimating transportation impacts, shipping distances from reactors to an AFR average 1000 miles for this generic statement. Shipping distances from reactors to a repository or from an AFR to a repository are assumed to average 1500 miles. Therefore, total shipping distance between a reactor and disposal can be as much as 2500 miles. Actual shipping distances would vary, of course, depending on sites selected.

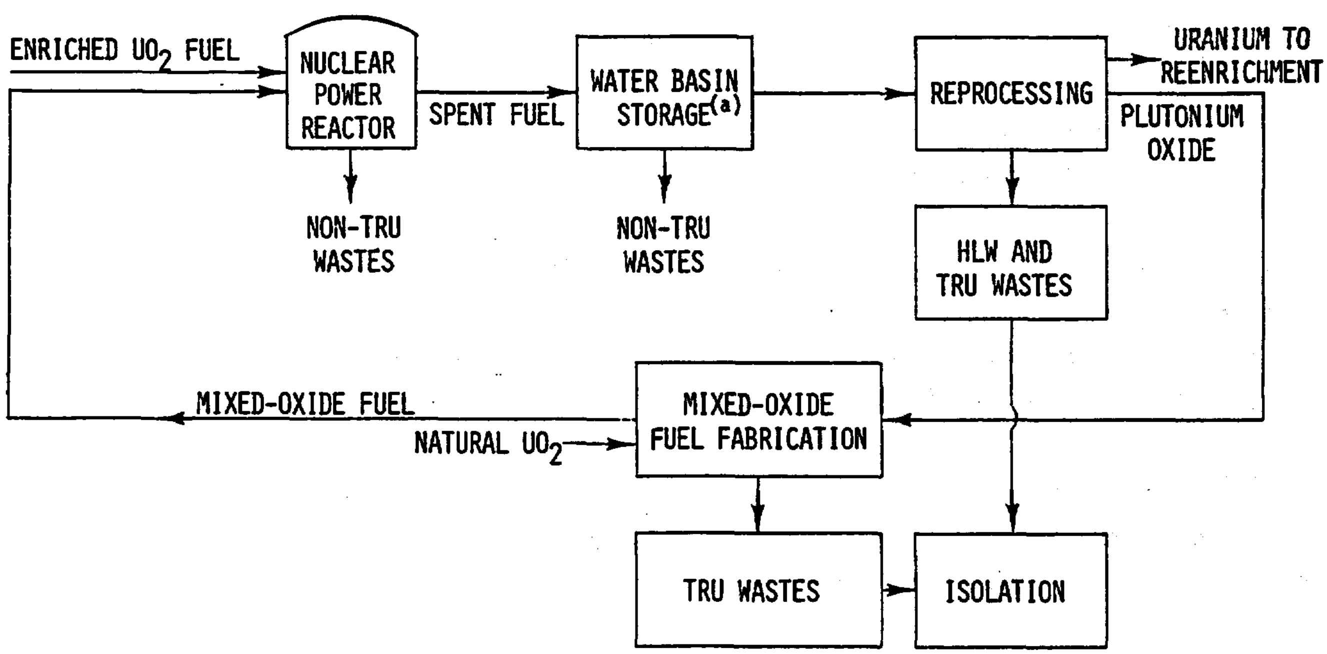
The logistics and storage requirements of this fuel cycle for several nuclear power growth assumptions are discussed in Chapter 7.

3.2.1.2 Reprocessing Fuel Cycle

A simplified diagram of the reprocessing fuel cycle is shown in Figure 3.2.2. In this fuel cycle, uranium and plutonium are separated from other components of the fuel and purified for recycle at a fuel reprocessing plant (FRP). The major process steps at the FRP, excluding waste treatment operations, which are described in Chapter 4, are:

- Underwater storage of spent fuel awaiting processing.
- Recovery and purification of the uranium and plutonium by solvent extraction using the Purex process. The reference plant, described in DOE/ET-0028, Section 3.2, operates 300 days per year to process 2000 MTHM/yr of spent fuel. The spent fuel elements are chopped into short sections so that the contained fuel can be dissolved in nitric acid. The uranium and plutonium are then extracted into an organic solvent phase containing tributyl phosphate (TBP), leaving the bulk of the fission products in the nitric acid solution (the high-level waste). The uranium and plutonium are separated and the remaining fission products removed in subsequent solvent-extraction process cycles.
- Conversion of plutonium to a solid at the FRP by precipitating plutonium as an oxalate, which is then separated and calcined to PuO_2.
- Conversion of the uranium from a nitrate solution to UF_6 at the FRP by calcining the uranium nitrate to UO_3, reducing the UO_3 to UO_2 with hydrogen, then converting the UO_2 to UF_4 by hydrofluorination with HF, and finally converting the UF_4 to UF_6 with fluorine. (UF_6 is the form required by the enrichment plant.)

Over 99% of the spent fuel fission products and about 0.5% of the uranium and plutonium would be contained in the FRP high-level waste. Substantial quantities of a variety of TRU



[a]WATER BASIN STORAGE IN EITHER REACTOR BASINS, AFR FACILITIES OR FRP BASINS

FIGURE 3.2.2 Uranium-Plutonium Recycle Fuel Cycle wastes also result. These are described more fully in Section 4.2. After the HLW is solidified (Section 4.3) it may be stored on-site (Section 4.4) for a period prior to shipment (Section 4.5).

A mixed-oxide fuel fabrication plant (MOX-FFP) prepares fuel containing a mixture of plutonium dioxide and uranium dioxide for recycle to a nuclear power plant. The reference MOX-FFP receives UO_2 and PuO_2 powders and Zircaloy cladding tubes and end plugs and prepares hermetically sealed fuel rods ready for insertion into fuel assemblies. The reference plant, described in DOE/ET-0028 Section 3.2, operates 300 days per year to produce 400 MTHM of LWR fuel/yr; up to 5% of the heavy metal content is plutonium. The major process steps involved include:

- Mechanical mixing of UO_2 and PuO_2 powders
- Preparation of dense fuel pellets by pressing, sintering, and grinding the mixed powder
- Sealing the pellets in Zircaloy cladding to form fuel elements
- Scrap recycle. The following assumptions are made about the reprocessing fuel cycle logistics:
- Spent fuel is stored until it is shipped to a reprocessing facility. As in the once-through cycle, storage can occur either at the reactor site or at an AFR. Reactor basin storage capacity is also seven annual discharges, but spent fuel is stored for a minimum of one year once this accumulated backlog of stored fuel is worked off. The reprocessing plant maintains a working inventory of 0.5-yr worth of spent fuel in storage. Thus, the minimum fuel age at reprocessing is 1.5 years; however, because a large accumulated inventory of spent fuel exists before the start of reprocessing, it is over 20 years after reprocessing starts before this minimum age is reached.
- The high-level waste is solidified immediately and then stored on-site for 5 years prior to shipment to a repository or to an interim storage facility if a repository is not available.
- TRU wastes are shipped immediately after treatment and packaging to either a repository or interim storage.
- Spent fuel shipping distances are assumed to average 1000 miles from reactors to an FRP or to an AFR, or from an AFR to an FRP.
- Treated waste shipping distances are assumed to average 1000 miles to interim storage and 1500 miles from either an FRP or from an interim storage facility to a repository. As in the once-through cycle, the actual distances will vary. No waste shipments between an FRP and a MOX-FFP are assumed.

The logistical and storage requirements of this fuel cycle as well as the once-through cycle for several nuclear power growth assumptions are discussed in Chapter 7.

3.2.2 Nuclear Power Growth Assumptions

To cover the range of potential waste management impacts in the years ahead, five different nuclear power growth scenarios are considered in this Statement.

A reference projection of 400 GWe of installed nuclear power capacity in the year 2000 and a bounding low projection of 255 GWe in the year 2000 was used in the original draft Statement (DOE/EIS-0046 D). Since that report was published for comments, however, studies (Clark and Reynolds 1979) conducted by DOE's Energy Information Administration (EIA) have indicated that the year 2000 installed nuclear power capacity is unlikely to exceed 250 GWe.[a] In addition, some comments on the draft Statement stated that the 400 GWe projection indicated a bias in favor of nuclear power development while other commenters objected that it overstated the magnitude of the waste management problem. For these reasons, the maximum projection for the year 2000 considered in this final Statement has been established as 250 GWe.

None of the projections or scenarios are intended to represent predictions of future developments. They are intended to encompass a possible range of nuclear power development and to provide a reasonable basis for estimates of waste management impacts as well as a basis for either interpolating waste management impacts to intermediate projections or for extrapolating waste management impacts to higher projected growth rates.

The waste management impacts for these scenarios are presented in Chapter 7.

The five scenarios are described below and the resulting nuclear power capacities are tabulated in Table 3.2.1 and plotted in Figure 3.2.3.

TABLE 3.2.1. Nuclear Power Capacity Assumptions, GWe

	Case 1 Present Inventory	Case 2 Present Capacity	Case 3 250 GWe in 2000 and Phaseout	Case 4 250 GWe in 2000 and Constant	Case 5 250 GWe in 2000 to 500 GWe in 2040
1980	50	50	55	55	55
1985	0	50	113	113	113
1990	0	50	155	155	155
1995	0	50	196	196	196
2000	0	50	250	250	250
2005	0	49	249	250	281
2010	0	44	244	250	312
2015	0	14	214	250	343
2020	0	0	195	250	374
2025	0	0	137	250	405
2030	0	0	95	250	437
2035	0	0	54	250	468
2040	0	0	0	250	500

(a) The referenced report did not project beyond 1995. The figure of 250 GWe in the year 2000 is based on an extrapolation.

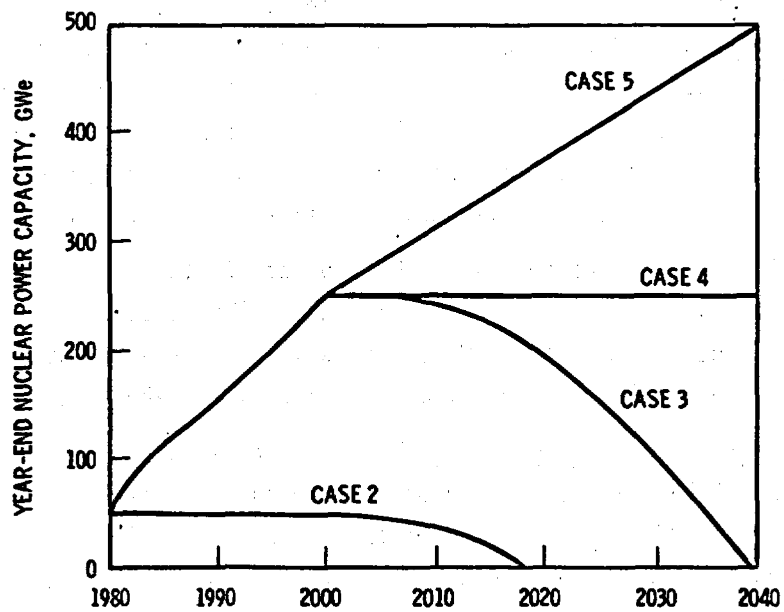


FIGURE 3.2.3. Nuclear Power Growth Assumptions

Case 1--Present Inventory--This case considers the requirements for management of approximately 10,000 MTHM of spent fuel that would remain if the 50 GWe of LWR capacity operating at the beginning of 1980 were shut down at the end of 1980 and all reactor cores discharged. However, no attempt is made in this Statement to consider or evaluate the broader issues of an industry shutdown (beyond those associated with handling the waste) such as national energy policy, impact on the economy, the impacts of alternative energy sources, costs, and the environmental impacts of such action.

Case 2--Present Capacity--This case considers the requirements for management of 48,000 MTHM of spent fuel that would result from continued operation of the existing 50 GWe of nuclear capacity to retirement after 40 years of operation with no further additions to this system. As in Case 1, no attempt is made to consider or evaluate the broader issues beyond the impact of handling the associated wastes, that would be involved in a limitation of this sort.

Case 3--250 GWe in Year 2000 and Phaseout--Case 3 assesses the waste management impacts for all aspects of a complete life cycle of a nuclear generating system including reactor shutdown, facility decommissioning, etc. In this case nuclear power capacity increases to 250 GWe in the year 2000. (This case follows the EIA high case projection through 1995.) After the year 2000, no additional nuclear power plant startups are considered. All nuclear power plants are assumed to operate for a 40-year life, after which they are decommissioned. Thus, the installed generating capacity of the system is reduced to zero in the year 2040. Based on average experience to date, average startup capacity factors of 59%, 63%, and 67% were assumed for the first three years of operation for all nuclear plants. Starting with the fourth year, each plant was assumed to operate at 70% for 22 years and then decline to 40% in its fortieth year after which it is shut down. A total of 239,000 MTHM of spent fuel is produced in this case.

We do not yet have sufficient operating experience with nuclear plants to predict this life cycle with high confidence. These plants are generally assumed to have lifetimes in the range of 30 to 40 years. The upper end of this range was used here to be conservative in regard to the amount of radioactive waste to be managed for a specific system. The declining load factor as facilities age has not yet been observed in nuclear plants but is similar to the experience of large central-station fossil-fuel generating units.

Using the year 2000 as a reference point, the impacts of other growth assumptions can be derived by comparison to this case. For example, a 500 GWe system in the year 2000 would produce approximately twice the impacts of Case 3 if allowed to run out its useful life, or a 125 GWe system in the year 2000 would produce approximately one-half as much impact.

Case 4--250 GWe in Year 2000 and Constant--This case follows the same growth pattern as Case 3 up to the year 2000. Then, instead of phasing out capacity as plants are decommissioned, new capacity is added to maintain the total capacity at 250 Gwe until the year 2040, beyond which time the case is not analyzed. A total of 316,000 MTHM of spent fuel is produced in this case.

This case illustrates the rate at which continuous waste management requirements and impacts would occur in a constant or steady-state system. An approximate equilibrium is established.

Waste management requirements and impacts at other constant capacity levels can be obtained by comparing capacities and impacts to this case.

Case 5--250 GWe in Year 2000 and 500 GWe in 2040--This case also follows the same growth pattern as Case 3 up to the year 2000. After that, however, capacity additions continue until a doubled capacity of 500 GWe is reached in the year 2040. Beyond the year 2040, the case is not analyzed. A total of 427,000 MTHM of spent fuel is produced in this case.

No equilibrium is established in this case. It illustrates the waste management requirements and impacts for a continuously expanding system. Results can be extrapolated to other growth rates by comparing the differences between the year 2040 capacities in Cases 4 and 5 to the difference in impacts. For example, a capacity of 750 GWe in the year 2040 would have twice the additional impact over Case 4 that Case 5 has.

3.2.3 Resource Commitment Assessment

In most instances, data describing environmental impacts that are caused by commitments of resources are presented as land and water requirements, material requirements, energy consumption, and manpower requirements for construction, operation, and decommissioning of the facilities. Resource commitments are combined by facilities on a single reference plant basis for analyzing predisposal activities in Section 4.7 and for geologic repositories in Section 5.4. Resource commitments are further aggregated by plant to systems of waste management and disposal within fuel cycle options in Chapter 7.

3.2.4 Ecological and Atmospheric Impacts

The impacts of the treatment, interim storage, transportation, and final disposal of radioactive wastes on natural ecosystems cannot be satisfactorily dealt with in detail in a generic sense because of the overriding influence of site-specific factors. For example, the expected impacts of certain waste technologies on plant and animal communities in an area of high precipitation may be markedly different from those in an arid environment. The ability of natural systems to withstand stress will vary widely according to their environment. Similarly, the economic worth of the natural resources at risk will depend greatly on the region and the degree of change already induced by human activities.

In this Statement, the assumption is made that environmental releases of radioactive wastes that are within the acceptable standards designed to protect man will also be within limits tolerable to natural plant and animal populations. In general, man is believed to be more sensitive to radiation than are other lifeforms. Thus, the discussion of potential radiation effects on plants and animals other than man is not considered on a generic basis. Consequently, discussion of the ecological impacts of radioactive waste management is confined mainly to 1) the effects on the use of land and surface water and 2) the impacts resulting from the release of nonradioactive chemicals and heat to the air and to surface water.

The main atmospheric effects evaluated in this Statement are the impacts on ambient air quality caused by emissions to the atmosphere during construction and operation of the facilities. Secondary emissions from construction force vehicles and construction equipment are also included in the emissions inventory. Since heat is a by-product of each process, its effect on the biosphere, whether released directly or via cooling tower, is also investigated.

3.2.5 Radiological Impacts Assessments and Uncertainties

Radiological impacts are probably perceived as the most important aspect of radioactive waste management. As a consequence, radiological aspects are considered in detail in this Statement and in its supporting documents. Radiological impacts are described principally in terms of dose to workers and to the public (The regional population is described in Appendix F; mathematical models are described in Appendix D.)

Doses to the public from waste management operations would be expected to arise from inhalation of radionuclides, by direct radiation, and from ingestion of food products (e.g., vegetables, meat, and dairy products) either grown on land contaminated by radionuclides deposited on the ground or contaminated by deposits directly on the food products themselves.

Dose from exposure to planned or unplanned releases of radionuclides to the biosphere is considered for three main categories of the public: the maximum individual,[a] the

(a) The maximum individual is a hypothetical resident whose habits would tend to maximize his dose.

population within a 50-mile radius reference environment of a waste facility (2 million), and the world population (6 billion in the year 2000).[a] In selected instances dose to the population of the eastern half of the United States is also presented.

Unless otherwise noted, doses are to the whole body; doses to other organs of interest are presented in DOE/ET-0029. Dose in this Statement is usually expressed as a 70-yr accumulated whole-body dose, although where informative, first-year doses are also given. In some instances, multigeneration doses are provided.

Health effects are calculated for regional or worldwide populations based on the dose received by these populations from the aggregation of the facilities involved. The doses calculated to result from individual facilities, except for nondesign basis repository accidents, are usually too small to warrant discussion of health effects.

In this Statement, 50 to 500 fatal cancers and 50 to 300 serious genetic defects are assumed to result in an exposed population for each million man-rem of radiation exposure received (for a total of 100 to 800 health effects per million man-rem). The possibility of zero risk is not excluded by the available data, i.e., there is a possibility that no cancers may be caused by low doses of radiation. For further discussion of the derivation of these risk factors, the reader should consult Appendix E.

Also presented is an alternative approach to analysis of exposure in which the estimated radiation doses from waste management activities are compared with more accurately known radiation doses from other sources such as naturally occurring radiation and radioactive materials.

Radiation dose calculations (Appendix D) use models to develop total doses by summing radiation doses from various radionuclides entering (or externally exposing) the human body. Each step in the dose calculation has uncertainty associated with it. A common radiation protection practice has been to assign values to parameters used in dose calculation that, if uncertain, will tend to overstate rather than understate the resulting dose.

3.2.6 Socioeconomic Impacts

The approach used in the analysis of socioeconomic impacts emphasizes changes in local employment and population caused by the construction and operation of a waste repository in selected geologic media. The repositories examined in this analysis generate socioeconomic impacts in several ways: through the employment requirements of construction and operation, through the demand generated for locally supplied materials and services, through secondary economic growth generated by the project, and through the public revenues resulting from project operation. In this generic Statement, the employment requirements are stressed because they more directly affect impacts (such as demands for housing, education, and health services) than do other requirements. Because tax structures and prospective revenues vary widely across potential sites no meaningful and representative estimates of reve-

(a) The only radionuclides that contribute significantly to worldwide radiation doses for the type of release mechanisms visualized here are ^{3}H, ^{14}C, and ^{85}Kr. For this reason, worldwide dose calculations are based on ^{3}H, ^{14}C, and ^{85}Kr only.

nue impacts can be provided in a generic study and no such estimates are prepared in this Statement.

A baseline population from the start of construction of a facility until scheduled decommissioning is projected. Work force requirements for the project are compared with the availability of workers already living in the area. Workers not available within commuting distance of the site will immigrate. The impact of their presence in the local area is increased to the extent that they either induce secondary growth in the local economy or bring family dependents with them. The total influx of new people to an area can equal three or four times the number of primary workers hired from outside the area. The model distributes the total new population to the site county and surrounding counties on the basis of county size, distance to the work site and availability of housing.

A generic assessment of the socioeconomic impacts incorporates the assumption that a variety of sites are potential candidates. Since the potential sites may differ considerably in terms of their distinguishing characteristics (especially population size, composition and distribution, industrial composition of the labor force, and availability of social services), the potential effects of project development on a number of alternative sites must be examined. In order to emphasize that the reference sites used in this analysis are hypothetical, they are simply labeled Midwest, Southeast, and Southwest. Each reference site consists of a single county. The region within which the county is located is defined as the aggregation of all counties falling substantially within a 50-mile radius of the site. The forecasting model allocates immigrants to these counties, then focuses upon the new population residing in the site county and upon the demands it places upon the county for social services. The objective of this generic analysis is to provide a range of probable socioeconomic impacts and to illustrate how variation in site characteristics and variations in construction and operating requirements with different disposal media combine to produce demographic and economic pressures upon local areas. Whether or not these pressures become translated into actual net socioeconomic impacts depends upon how each community responds in terms of the capacity of the service system to absorb new demands, the willingness of the community to adjust to pressure for change, and the availability of mitigating strategies to the community.

3.2.7 Basis for Accident Analysis

The accident analysis procedure for this Statement involves several steps. First, potential accidents are identified for each waste management function and alternative technology. Next, accidents are divided into four categories based on considerations of their potential to expose plant workers to significant radiation levels and/or release radioactive material to the environment. Accidents in each severity category are then grouped by similar release characteristics. Finally, the largest potential accident release category/accident severity group is selected for environmental consequence analysis. In all, 207 possible accident types were examined for the waste management system with 116 of these having potential for offsite releases of radioactive material. Forty-six (46) of the releases were analyzed for environmental impacts.

A listing of all accidents considered in this analysis and the grouping of releases to determine source terms for environmental consequence analysis is given in Section 3.7 of Technology for Commercial Radioactive Waste Management (DOE/ET-0028). Environmental impacts of specific source terms are presented in the Environmental Aspects of Commercial Radioactive Waste Management (DOE/ET-0029).

Each waste management technology was examined for potential accidents which might result in offsite releases or significant impact on plant operations. Potential hazardous material releases (called source terms) were developed for these accidents using successive release fractions. The release fraction is the fraction of radionuclide inventory that is released to the next containment barrier or to the environment. The radioactivity released in an accident may be substantially reduced by one or more barriers, such as high-efficiency particulate air (HEPA) filter banks. The radioactivity released to the environment was obtained by multiplying the product of the release fraction for each release mechanism and containment barrier (e.g., the accident, process equipment, HEPA filters, etc.) by the radionuclide inventories involved in the operation. Where more than one waste management technique was examined, analysis was based on the example system waste form (see figure 4.1.3 on page 4.8 for the identification of the example waste forms).

Accident frequency estimates were developed where possible. In the absence of actual accident experience estimates are based on previous experience with similar equipment, while others are engineering judgment based on review of the conceptual designs.

Following source term and frequency definition, the lists of representative accident scenarios were classified into three accident severity groups:

1. Minor--Process interruptions without potential for significant release of radioactive or other hazardous materials.
2. Moderate--Events with potential for small radioactivity release.
3. Severe--Events with a potential for significant radiation hazards.

The three accident classifications cover the spectrum of design-basis accidents. Nondesign-basis accidents (a fourth category) includes all accidents which exceed site criteria[a] (e.g., meteorite impact) or involve concurrent independent failure of process and multiple containment system barriers. By virtue of plant design and operational techniques, the possibility of nondesign-basis accidents is extremely unlikely during the design life of the waste treatment or storage facility and are not considered for these facilities. However, for geologic isolation, because of the long period of required containment, several nondesign-basis accidents (or unexpected events) are postulated (Section 5.5).

An umbrella source term concept was used to limit the number of accidents requiring detailed impact analysis. Viewed independently of accident initiation sequences and fre-

(a) Site criteria include: 1) definition of the maximum credible earthquake, surface faulting, floods and wind velocities based on historical evidence, local and regional geology, and expert judgment; 2) local and regional demography; and 3) proximity and definition of hazards caused by man.

quencies, source terms can be grouped by release severity for environmental consequence analyses. Releases were classified based on similar release pathways, chemical form, accident severity category, and isotope types released (fission products, activation products, and actinides). The largest release from any of the accidents in a similar release group was selected as the umbrella source term for that group. A summary description of impacts from the umbrella source terms for each waste management step is presented in Sections 4.8 and 5.4.

Releases of radioactive material to the environment result from both accidents and normal operational releases. Operational releases result from routine handling or processing of radioactive materials and are limited by the containment system design and performance. They are expected to occur at a relatively uniform rate over the life of the plant. Accidental releases occur intermittently because of operational error or because of system component or containment failures. Severity of releases is generally inversely proportional to their frequency. The small-release, moderate-frequency minor accidents were characterized for impact analysis in two ways: 1) as short-term intermittent release to describe their accidental nature and 2) as integrated releases averaged over one year to describe their moderate frequencies of occurrence. Integrated annual releases caused by minor accidents were added to facility releases from normal operations in determining environmental impacts for normal operation. Because of their low frequency, releases from moderate and severe accidents are described as separate impacts and are not included in consequences of routine operation.

3.2.8 Cost Analysis Bases

Estimates of capital and operating costs for waste management predisposal operations and disposal in geologic repositories were developed for this Statement. This section summarizes the assumptions and methodology used to derive these cost estimates, as well as the bases for estimating uncertainty ranges. A complete discussion of cost bases and assumptions is given in DOE/ET-0028, Vol. 1, Section 3.8.

The cost estimates themselves are summarized in Sections 4.9 and 5.6 for predisposal and geologic-isolation operations, respectively. Additional cost information on other disposal alternatives where the data base is generally more limited, is presented in the individual discussions of these alternatives in Chapter 6. An analysis of the overall systems costs of waste management and their impact on the cost of electric power is given in Chapter 7. The costs presented in Chapter 7 represent a full cost recovery of all identifiable costs including R&D costs and government overheads.

3.2.8.1 Bases for Capital, Operating and Decommissioning Cost Estimates

A constant dollar method of analysis is employed in which all costs, both present and future, are expressed in terms of the buying power of the dollar in mid-1978.[a] This is

(a) The costs from DOE/ET-0028 were originally derived in terms of 1976 dollars and have been escalated here to 1978 dollars by multiplying by 1.17. 1980 dollar costs can be approximated by multiplying by 1.20.

not meant to imply that inflation will not occur; rather, cost relationships can be more easily understood and placed in perspective if they are stated in constant dollar terms. Over the long term, the estimated costs developed in this study will increase at a rate comparable to the general rate of inflation.

Capital costs were derived by estimating requirements for major equipment, buildings and structures, site improvements, and construction labor. Factors were then applied to these direct cost estimates to generate other direct costs, indirect costs, architect-engineer costs, owner's staff costs during construction, initial inventory costs and other startup costs.

Operating costs include all cost items identified with operation. The number of man-hours, quantities of materials, and requirements for utilities were derived in each case from the facility descriptions. The allowances for maintenance, overhead, and miscellaneous costs were derived by applying factors to either capital or direct labor costs.

The capital and operating cost methodology outlined above is used to estimate all of the costs given in this Statement except for those of the transportation facilities (cost development for transportation is discussed separately in Subsection 3.2.8.4). An allowance for working capital is also provided. Working capital is defined as the cash required to operate a facility, i.e., the difference between current assets and current liabilities. This cash is treated as an outflow of funds during the first year of plant operation and as an inflow during the last year of operation. Working capital requirements are estimated at 50% of the first year's operating cost.

The cost of waste management in this Statement also includes the cost of facility decommissioning. Specific cost estimates were developed for decommissioning a reference spent fuel storage facility, mixed oxide fuel fabrication plant, and fuel reprocessing plant. Based on these estimates, the costs to decommission individual waste management facilities not otherwise included in the decommissioning of these primary facilities were estimated at 10% of their capital costs (except for underground repository facilities for which separate estimates were made). These costs are incorporated in the levelized unit cost calculations for these waste management facilities. The costs of decommissioning FRP and MOX-FFP facilities are included in the waste management system costs (Section 7.6).

3.2.8.2 Bases for Levelized Unit Cost Estimates

Levelized unit costs are capital and operating costs translated into equivalent, constant (or level) annual unit costs. The unit cost is sufficient to pay any interest charges on debt; pay all operating expenses, taxes and insurance; earn a specified return on outstanding capital; and recover the capital investment over the life of the project. In summary form the levelized unit cost relationship can be expressed as:

$$\text{Levelized Unit Cost} = \frac{\text{Annualized Capital and Operating Costs}}{\text{Annualized Units Processed}}$$

Since the calculated unit costs are a function of taxes and returns on equity and debt, ownership for each facility is defined as either private industry, Federal, or utility ownership. The constant dollar weighted average cost-of-money rates and ranges (excluding an inflation premium) used in the levelized unit cost estimates are 10 ± 4%, 7 ± 3%[a] and 7 ± 2% for private industry, Federal, and utility ownership, respectively. Also included in the unit cost calculations are property taxes and state income taxes as well as Federal income taxes, accident and hazard insurance, and investment credits.

For this Statement, most unit costs are based on a 15-yr economic plant life. The text notes when plant lives other than 15 years are used, as in some of the storage facilities. However, because of the cost-of-money effect over long time periods at the rates employed here, plant lives longer than 15 years have only a small effect on unit costs. Although it is not anticipated, the entire facility could be replaced after 15 years with no increase in unit costs (in constant dollars) beyond those estimated here.

3.2.8.3 <u>Uncertainty Ranges for Cost Calculations</u>

Uncertainties in the levelized unit cost estimates were derived from uncertainties calculated for three components: 1) capital costs, 2) operating costs, and 3) the cost of money. The range for capital costs reflects uncertainties in the definition of the engineering scope required to provide a fully-functional plant based on the technology described, as well as uncertainties in the pricing and quantities for labor, materials, and equipment. A contingency covering these and similar factors has been included in the base capital cost estimate. The uncertainty for capital costs ranges from about ±20% to ±45%, depending on the facility and equipment, with a median uncertainty of about ±30%. The uncertainty in the operating costs for most facilities is estimated to range from +50% to -25%.

Because of the capital-intensive nature of the nuclear industry, the dollar value of the capital charge uncertainty generally overshadows the dollar value of the operating cost uncertainty for most of the facilities evaluated. A weighted overall uncertainty range was calculated for each unit cost based on the three component uncertainties. A statistical analysis of several example unit cost calculations, assuming a normal random distribution of uncertainty around the three variables, indicates that there is a 95+% probability of being within the total uncertainty range cited for each levelized unit cost.

3.2.8.4 <u>Cost Estimates for Transportation</u>

The unit cost development for waste transport was somewhat different than for other waste management facilities.

Estimates of capital costs of transportation equipment were made assuming the equipment is supplied repetitively by qualified vendors on a competitive basis. The capital cost estimate covers costs for the complete transportation system including the cost of the cask,

(a) Use of the 7% cost of money or discount rate for a Federal project is based on the assumption that a full cost recovery methodology would be adopted similar to that described in DOE/EIS-0015, Vol. 4., where possible charges for AFR storage of spent fuel are described and a 6.5% discount rate is employed. The ±3% range encompasses the 10% rate specified in the 1972 OMB circular No. A-94 for use in evaluating government projects. The basis for the private industry and utility discount rates is described in DOE/ET-0028, Vol. 1.

rail car or truck trailer, tiedown system, cooling equipment (if needed), and sun shields. Costs of locomotives and tractors were included in the freight or haulage charges and costs of the waste containers were included in the predisposal waste treatment costs.

The capital costs were translated into unit cask use charges, using the unit cost calculational procedure, private ownership financial parameters and the cask capacity. A cask use factor of 80% (292 days per year) and an annual maintenance charge of 2% of the capital costs were assumed.

Round-trip freight or haulage charges were developed (see DOE/ET-0028, Vol. 4, Section 6) for both rail and truck transportation. A unit freight charge was developed by dividing the freight charge per trip by the cask capacity. The total unit transport cost was obtained by adding the unit cask use charge to the unit freight charge. Additional detail on transportation cost calculations is given in the previously mentioned reference.

3.2.8.5 Research and Development Costs

Costs for research and development have been included in the overall systems costs for waste management developed in Chapter 7.

3.2.9 Physical Protection Safeguard Requirements Assessment

The characteristics of spent fuel, the waste materials and the facilities were reviewed and safeguard requirements were identified for each of the waste management steps considered in this Statement. Results of this assessment are summarized in Section 4.10 for predisposal activities, in Section 5.7 for mined geologic repositories and in Section 6.1 for other disposal alternatives.

Safeguard requirements for plants and materials in the nuclear industry are specified in the Code of Federal Regulations (10 CFR 70 and 10 CFR 73). They include physical protection measures employed to prevent the theft or diversion of special nuclear material, to prevent the willful release of radioactive material, and to prevent the sabotage of nuclear facilities. The principal features of these requirements (10 CFR 73) are the protection forces (guards), physical and procedural access controls, intrusion detection aids, communications systems, and plans for emergencies and strict accountability (10 CFR 70) of all items containing nuclear material including fuel elements and containers of waste. Equipment items, systems, devices, or materials whose failure, destruction or release could directly endanger the public health and safety by exposure to radiation are defined as "vital" (10 CFR 73). Under the existing Code of Federal Regulations, spent fuel and some waste materials in the reprocessing cycle would be classified as vital, and the areas in which they are processed would be vital areas. As such, these areas would require substantial levels of physical protection. For example, Federal regulations specify two independent and successive physical controls over personnel and vehicular entry and exit to and from vital areas.

The required physical protection measures are affected by the potential risk of theft of material that has special strategic worth or is highly radioactive, or by the consequences to the public following sabotage at a facility handling these materials. The level of the potential risk will in turn be determined by the characteristics of these possible targets and the kind and degree of threat anticipated.

Safeguard requirements for the waste management facilities considered in this Statement were characterized based on the attractiveness and accessibility of the wastes as potential targets for theft or sabotage. Attractiveness depends on composition and physical form of the waste. The important aspects of composition are the concentration of fissionable materials and radioactivity. Radioactive wastes are not considered good sources of fissile material for the manufacture of a weapon because of the small quantities of fissile materials per unit volume. Of the waste forms considered in this Statement, only spent fuel contains attractive quantities of such materials. However, the physical condition of spent fuel waste requires sophisticated processing in order to recover the fissile material. Some highly radioactive nuclear wastes may be in a form that would be attractive to an adversary as a source of material that is readily dispersable and, because of the health hazard, could be used to threaten and extort gains from industries or public agencies.

In evaluating the potential for sabotage, consideration was given to design features that could significantly reduce the consequences of sabotage and contribute to the protection of this material. These design features include the thick shielding around the more radioactive process vessels (walls up to 2 m thick); tornado, earthquake and flood protection requirements for all key process facilities; monitored cells and operations; and equipment for detecting and coping with releases of radioactivity. These features generally result in facilities that are unattractive targets for sabotage.

Accessibility of the waste materials was also considered. Factors affecting accessibility include: 1) quantity available at a given location, 2) the degree of isolation of the location, and 3) the complexity of the devices necessary for handling the material (e.g., whether they are operated manually or automatically and whether special knowledge or skills are required).

The final element considered in assessing safeguard requirements was the threat level of potential adversaries. The overall safeguard risk was assessed by considering the above elements--the attractiveness of the material, its accessibility, and the threat level--in the following relationship:

Risk to Society = Frequency x Success Rate x Consequences

The frequency of attempts, related in part to the attractiveness of material; the success rate, related in part to the availability of the material; and the consequences, measured by effects on the public and the environment, are also all affected by the skills, motivation, financial backing and intrepidness of potential adversaries. All contribute to the risk to society. The relationship shows that if one or more of these factors is very small, the risk to society is also small.

Frequency and success probabilities are difficult to define. However, safeguards measures normally in place for the vital facilities and vital materials of the fuel cycle are designed to reduce the frequency and success rate to very small values. The safeguard measures will also significantly reduce the consequences of an adverse action through implementation of safeguard emergency plans by providing effective response to threats and attempted adversary actions, and by providing effective assistance to public agencies in protecting the public from the consequences of these threats and actions.[a]

(a) See Appendix E of 10 CFR 50 and Appendix C of 10 CFR 73.

REFERENCES FOR SECTION 3.2

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3.3 NATURALLY OCCURRING RADIATION AND STANDARDS FOR EXPOSURE TO MAN-MADE RADIATION

Although public awareness regarding radiation has grown markedly in recent years, many readers may not be aware of all of the kinds and quantitites of naturally occurring radiation around them. Because of this and because naturally occurring radiation can often be used as a meaningful perspective for evaluating radiation exposure from other sources, a summary of radiation from naturally occurring sources is provided.

To protect workers and the public from excessive exposure to man-made radiation sources and yet realize the benefit from the use of these radiation sources, standards or limits of exposure for various circumstances have been established by several authoritative bodies. Exposures up to these standards are believed not to result in undue risk to the individual. Regardless, the practice of keeping exposures as low as reasonably achievable is fundamental in the radiation protection field. As a consequence, in many facilities the average exposure is not more than one-tenth of the occupational standard. Because of the importance of standards in the control of radiation exposure, a summary of presently applicable standards is also presented.

3.3.1 Natural Radioactivity and Radiation Dose[a]

Depending on their activities and location, people are exposed in varying degrees to several sources of ionizing radiation found in nature. Cosmic radiation entering the earth's atmosphere and crust is one natural source of exposure. Also, nuclear interactions of cosmic rays with matter produce radiation and radionuclides to which people are exposed. Other sources exposing people to radiation are naturally occurring radioelements in the earth's crust.

Natural radioactivity includes all ionizing radiations and radionuclides except those that have been produced by man's activities, such as that produced by nuclear weapons, bombardment of targets by ion accelerator beams, in nuclear reactors, and from medical and dental x-rays. Sometimes a distinction is made between natural radioactivity in an unmined uranium ore body and "enhanced radioactivity" in mine or mill tailings, for example, radioactivity left on the earth's surface.

The following discussion of dose[b] and dose rate to the U.S. population from natural radioactivity is presented as perspective for dose estimates associated with management of commercial radioactive wastes in the LWR fuel cycles. No contention is made that exposure to natural radioactivity is or is not harmful. However, when doses associated with waste management are small fractions of natural background dose, such doses would probably be viewed as insignificant.

(a) The discussion of natural radioactivity was taken largely from Natural Background Radiation in the United States, NCRP Report No. 45, Washington, DC, 1975.
(b) Throughout this Statement, the term "dose" may generally be taken to mean the more rigorous term "dose-equivalent." The latter; expressed in units of rem or millirem (one one-thousandth of a rem), implies a consistent basis for estimates of consequential health risk, regardless of rate, quantity, source, or quality of the radiation exposure. Unless otherwise specified, dose is that for the whole body.

3.3.1.1 Cosmic Radiation

Cosmic radiation refers both to primary energetic particles of extraterrestrial origin that strike the earth's atmosphere and to secondary particles generated by the interaction of primary particles with the atmosphere (radionuclides produced by cosmic radiation are discussed later). The primary cosmic radiation consists of particles produced outside the solar system and particles emitted by the sun. The cosmic ray dose rate to the population living at sea level is about 26 mrem per year, taking into account shielding from structures. Considering the altitude distribution of the U.S. population, the average dose rate is 28 mrem per year. In Denver, which is the largest city at a relatively high altitude (1600 meters) in the United States, the average dose rate from cosmic rays is about 50 mrem per year. In Leadville, Colorado (3200 meters), which has a population of about 10,000, the average cosmic ray dose rate amounts to 125 mrem per year. High altitude airplane flights add a small fraction to the population dose from cosmic rays at ground level. For example, a jet flight of 5 hours duration (e.g., transcontinental or transatlantic at 12 km altitude) at mid-latitudes would result in a dose of approximately 2.5 mrem to the whole body. An extreme case would be a 10-hr polar route flight from, for example, California to Europe where the long flight time and the higher cosmic ray intensities at high latitudes would result in a passenger dose of approximately 10 mrem (or 20 mrem for a round trip).

3.3.1.2 Terrestrial Radioactivity

Terrestrial radioactive material is present in the environment because naturally radioactive isotopes are constituents of a number of elements in the earth's crust. The nuclear interaction of cosmic rays with constituents of the atmosphere, soil, and water also produce a number of different radionuclides. These naturally occurring radionuclides give rise to both external and internal irradiation of man.

Cosmogenic Radionuclides

Cosmogenic radionuclides are produced through interaction of cosmic rays with atoms in the atmosphere and in the outermost layer of the earth's crust. The entire geosphere contains radionuclides produced in this fashion. The four cosmogenic radionuclides that contribute measurable dose to man are hydrogen-3 (tritium) (^{3}H), beryllium-7 (^{7}Be), carbon-14 (^{14}C), and sodium-22 (^{22}Na), all produced in the atmosphere. The total contribution to the average dose rate (in addition to direct cosmic radiation) by these four nuclides is less than 1 mrem/yr.

Primordial Radionuclides

Several dozen naturally occurring nuclides are radioactive with half-lives of at least the same order of magnitude as the estimated age of the earth (4.5×10^{9} yr), and are consequently assumed to represent a primordial inventory (that is, some radionuclides are remaining since the formation of the world). There are three chains or series radionuclides headed by thorium-232 (^{232}Th), uranium-235 (^{235}U), and uranium-238 (^{238}U). These radionuclides decay ultimately to a stable isotope of lead through a chain of decaying nuclides of wide ranging half-lives. These chains contain the, perhaps more familiar, nuclides radium-226 (^{226}Ra) and radon-222 (^{222}Rn) as well as 31 other radionuclides. Other radionuclides decay directly to stable nuclides. The most significant of the primoradial radionuclides in terms of dose is potassium-40 (^{40}K). Aside from a small contribution to dose by rubidium-87 (^{87}Rb), the remainder of the primordial radionuclides, including plutonium-244 (^{244}Pu), occur in extremely small amounts and make no significant contribution to dose. Doses resulting from these primordial radionuclides are discussed below.

External Gamma Radiation. The significant contributors to dose to people from outside of their bodies are ^{40}K and the decay products of the ^{238}U and ^{232}Th series. The principal determinant of outdoor terrestrial radiation at a given location is the soil concentration of natural radionuclides. In addition to soil composition, the radiation outdoors varies depending on the moisture content of the soil, the presence and amount of snow cover, and on the radionuclide concentration in the atmosphere which itself is quite variable. Indoors, the level of radiation is modified by the degree of shielding provided by the building materials against the outdoor radiation, and the amount of radiation originating from radionuclides in the building materials. Variations in outdoor radiation will be partially reflected indoors and, in addition, the contribution from radon decay products will depend on the room air ventilation rate. Each of these factors can play an important role in determining the exposure received by the population.

The overall population-weighted dose rate in the United States from external terrestrial radiation is estimated to be 28 mrem/yr. Moreover, variability in external terrestrial radiation is larger than that for other natural sources of human exposure. This variation in dose rate is characterized by nominal external terrestrial dose rates to the whole body of 15, 30, and 55 mrem/yr for the Atlantic and Gulf Coastal Plains, for the majority of the United States, and for an undetermined area along the Rocky Mountains, respectively.

Internally Deposited Radionuclides. While all natural radionuclides may add to internal (inside the body) radiation doses, only a few are found to be significant contributors. These include ^{3}H, ^{14}C, ^{40}K, and ^{226}Ra and ^{228}Ra and their decay products. Within the United States, all of these are relatively uniformly distributed so that their levels in foods and water do not vary appreciably with geographic location. In the United States widespread food processing and widespread transportation of foods and people have an additional "averaging" effect on radionuclide contents of diets throughout all geographic areas.

The average total internal whole-body dose rate of about 22 mrem/yr is dominated by about 20 mrem/yr from ^{40}K.[a] Dose rates to specific organs from internally deposited radionuclides are about 30 mrem/yr to the gonads and other soft tissues, 60 mrem/yr to bone

(a) Potassium is an essential element in the body and is physiologically controlled, hence variations in dietary composition will have little effect on body content or radiation dose received. The same is largely true for the cosmogenic radionuclides ^{3}H and ^{14}C.

surfaces, and 25 mrem/yr to bone marrow. The dose to women from internally deposited radionuclides is about 25% lower than that to men, because of their smaller potassium content per unit body weight.

Dose to Lung from Inhaled Radionuclides. Dose to the lung from natural airborne radionuclides results principally from the alpha-emitting daughters of ^{222}Rn. The short range of alpha radiation means that the doses are delivered locally to the lung tissue, particularly to the bronchial epithelium. The average dose rate to the total lung is about 90 mrem/yr, while the bronchi epithelium receives about 450 mrem/yr.

Variability in dose rate to the lung is dependent on local concentrations of ^{222}Rn. There is some increase in areas with elevated levels of ^{238}U and ^{226}Ra in soil and a decrease in coastal regions during periods of onshore winds. Levels of ^{222}Rn indoors are dependent on the building's structural materials and ventilation rates. Dose rates to the lungs of smokers from the long-lived decay products lead-210 (^{210}Pb) and ^{210}Po from ^{222}Rn may be up to three times higher than for nonsmokers.

3.3.1.3 Summary of Whole-Body Dose

From the foregoing, the combined whole-body dose rates from terrestrial radioactivity received by groups at 1) sea level for the Atlantic and Gulf Coastal Plains, 2) for the majority of the United States, and 3) for an undetermined area along the Rocky Mountains is 15, 30, and 55 mrem/yr, respectively. The internal and cosmic ray dose rate to the whole body adds about 50 mrem/yr, which results in totals of 65, 80, and 105 mrem/yr as shown in Table 3.3.1.

The whole-body dose rate for groups living at an altitude of 1500 m would be increased by about 20 mrem/yr from the increased cosmic ray radiation. A total whole-body dose rate of 125 mrem/yr from all sources essentially represents the situation for the city of Denver, where both cosmic and terrestrial components are higher than average.

In this Statement, doses calculated as resulting from various waste management activities are often compared with the dose received from naturally occurring sources. To avoid use of ranges of naturally produced doses and to suggest the lack of certainty in the value for any individual, a well-rounded 100 mrem/yr dose rate has been used for illustration. On that basis, the doses used in this report for the population and time periods cited are as given in Table 3.3.2.

TABLE 3.3.1. Summary of Average Whole-Body Dose-Equivalent Rates from Naturally Occurring Radiation, mrem/yr

	Cosmic Rays (Sea Level)	Terrestrial Radiation External	Terrestrial Radiation Internal	Total
Atlantic and Gulf Coastal Plains	28	15	22	65
Majority of U.S.	28	30	22	80
Rock Mtn. Area	28	55	22	105

TABLE 3.3.2. Nominal Whole-Body Dose Equivalents from Naturally Occurring Radiation

	Annual Dose	70-Year Accumulated Dose
Individual	0.1 rem	7 rem
Regional Population (2 million)	2×10^5 man-rem[a]	1.4×10^7 man-rem
World-Wide Population (6 billion)	6×10^8 man-rem	4×10^{10} man-rem

(a) Man-rem: the sum of the product of the dose received and the number of individuals receiving that dose.

Using the foregoing population doses from naturally occurring radiation and the relationship between population dose and health effects as described in Appendix E (50 to 500 fatal cancers plus 50 to 300 serious genetic defects per million man-rem),[a] the number of health effects that might be associated with naturally occurring radiation were calculated and are presented in Table 3.3.3.

TABLE 3.3.3. Health Effects Calculated for 70-yr Accumulated Dose from Naturally Occurring Radioactive Sources

	Fatal Cancers	Serious Genetic Defects	Total Health Effects
Regional Population (2 million)	700 to 7,000	700 to 4,000	1,400 to 11,000
World-Wide Population (6 billion)	2,000,000 to 20,000,000	2,000,000 to 10,000,000	4,000,000 to 30,000,000

3.3.2 Applicable Standards for Radiation Exposure Control

A number of existing standards provide for administrative control of potential radiological impacts from waste management operations. These are embodied either in the Code of Federal Regulations (CFR) or comparable codes of state and local governments. Some of these standards are presented here and a more extensive treatment is given in Appendix C.

3.3.2.1 Basic Radiation Standards

The basic radiation standards that apply to all NRC licensees are given in Title 10

(a) Other suggested conversion factors would indicate more effects and others less, not excluding zero effects. The Committee on the Biological Effects of Ionizing Radiation (BEIR), National Academy of Sciences, released in July of 1980 an updated report, the BEIR III report, that indicates risk estimates of cancer death from low levels of radiation are only half what they were thought to be eight years ago (as reported in the BEIR I report, 1972). The range of conversion factors used in this statement encompass the values suggested in both the BEIR I (1972) and BEIR III (1980) reports.

Part 20 of the Code of Federal Regulations (10 CFR 20). Title 10 is based on NCRP, ICRP and FRC guidelines (25 F.R. 4402 et seq May 18, 1960) on radiation standards and the U.S. Government has endorsed the model regulatory code of the United Nations, which closely follows ICRP philosophy. An excerpt from 10 CFR 20 follows:

> 20.101 Exposure of individuals to radiation in restricted areas.* (a) Except as provided in paragraph (b) of this section, no licensee shall posses, use, or transfer licensed material in such a manner as to cause any individual in a restricted area to receive in any period of one calendar quarter from radioactive material and other sources of radiaton in the licensee's possession a dose in excess of the limits specified in the following table:

	rem/calendar quarter	(rem/year)
Whole body; head and trunk, active blood forming organs; lens of eyes, and gonads	1-1/4	(5)
Hands and forearms; feet and ankles	18-3/4	(75)
Skin of whole body	7-1/2	(30)

> (b) A licensee may permit an individual in a restricted area to receive a dose to the whole body greater than that permitted under paragraph (a) of this section, provided:
>
> (1) during any calendar quarter the dose to the whole body from radioactive material and other sources of radiation in the licensee's possession shall not exceed 3 rems; and
>
> (2) the dose to the whole body, when added to the accumulated occupational dose to the whole body, shall not exceed 5 (N-18) rems where "N" equals the individual's age in years at his last birthday.

*"Restricted Area" means any area whose access is controlled by the licensee to protect individuals from exposure to radiation and radioactive materials.

Title 10 Part 20 also tabulates limiting concentrations in air and water for many radionuclides, for both the working environment and unrestricted areas, which are not to be exceeded. For individuals in restricted areas, these concentration limits have been calculated, based on continuing exposure for 50 years and standard physiological parameters, to give doses no higher than either those specified above or 15 rem per year to non-specified organs of the body.

For unrestricted areas, standards specify that no individual should receive a dose to the whole body in any one calendar year in excess of 0.5 rem, although some exceptions based on primary concurrent limits (see 10 CFR 20.105) do allow higher doses. In addition, the average dose from all modes of exposure to "a suitable sample of an exposed population group" should not exceed one-third of the limiting dose criteria. Concentration Guides for air and water in unrestricted areas are based on limits of the resultant annual dose to individuals (to either the whole body or specific body organs) of not more than one-tenth the limiting dose for restricted areas.

Since radiation protection guides for the general public are based on averages over a period of 1 year or longer, the evaluation of long-term average exposures should include consideration of reasonable annual occupancy factors as well as the variability of the exposure rates.

3.3.2.2 Other Requirements

EPA Uranium Fuel Cycle Standards

Federal Reorganization Plan No. 3 of 1970 specifically transferred to the Environmental Protection Agency (EPA) the authority to establish standards for "quantities of radioactive materials in the environment." Under this authority, EPA in 1977 issued regulations (40 CFR 190) prescribing "Environmental Radiation Protection Standards for Nuclear Power Operations," which read in part:

> 190.02 Definitions
>
> (b) "Uranium fuel cycle" means the operations of milling of uranium ore, chemical conversion of uranium fuel, generation of electricity by a light-water-cooled nuclear power plant using uranium fuel, and reprocessing of spent uranium fuel, to the extent that these directly support the production of electrical power for public disposal sites, transportation of any radioactive materials in support of these operations, and the reuse of recovered non-uranium special nuclear and by-product materials from the cycle.
>
> 190.10 Standards for Normal Operations
>
> Operations covered by this Subpart shall be conducted in such a manner as to provide reasonable assurance that:
>
> (a) the annual dose equivalent does not exceed 25 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems to any other organ of any member of the public as the result of exposures to planned discharges of radioactive materials, radon and its daughters excepted, to the general environment from uranium fuel cycle operations and to radiation from these operations.
>
> (b) the total quantity of radioactive materials entering the general environment from the entire uranium fuel cycle, per gigawatt-year of electrical energy produced by the fuel cycle, contains less than 50,000 curies of krypton-85, 0.5 millicuries of iodine-129, and 0.5 millicuries combined of plutonium-239 and other alpha-emitting transuranic radionuclides with half-lives greater than one year.

By definition these regulations do not apply to transportation or operations at waste disposal sites but do apply to reprocessing of spent uranium fuel for reuse in the generation of electricity.[a] Where applicable these regulations supersede the related portions of 10 CFR 20. The basis for the numerical values given was a cost/benefit analyses of expected reductions of estimated environmental doses and consequent "health effects" versus estimated dollar costs of additional effluent treatments.

Clean Air Act Amendments of 1977

The 1977 amendments to the Clean Air Act specifically required the EPA Administrator to determine whether emissions of radioactive pollutants will cause or contribute to air pollution which may endanger public health. The Administrator has made an affirmative

(a) EPA is presently developing radiation protection standards for the disposal of high-level waste. In addition, NRC has published an Advanced Notice of Proposed Rulemaking relative to their technical criteria for geologic disposal of high-level waste.

finding and listed radionuclides as hazardous air pollutants under Section 112 of the Act (44 FR 76738, December 27, 1979). EPA must now propose regulations establishing emission standards for radionuclides.

Marine Protection, Research and Sanctuaries Act of 1972. Public Law 92-532

Dumping of any material into ocean waters is permitted only pursuant to a permit from EPA, or, for dredged material, the Corps of Engineers. The Act specifically precludes issuance of a permit for dumping of high-level radioactive waste.

Department of Energy Requirements

Other than the quarterly fractionation of the Nuclear Regulatory Commission dose limits, and with minor exceptions for specific body organs, the limiting dose criteria of 10 CFR 20 are the same for Department of Energy operations, as given in ERDA Manual Chapter 0524 (ERDA 1975). Any new facilities for commercial high-level waste management are expected to be licensed by the Nuclear Regulatory Commission.

State Regulations

Under Section 274 of the Atomic Energy Act of 1954 as amended, a number of states and the Nuclear Regulatory Commission have executed agreements that permit a state to grant licenses for the control of specified nuclear activities within the state boundaries. Production and utilization of special nuclear materials and Federal facilities are specifically excluded. Examples of state-licensed activities are the commercially operated low-level waste burial sites at Barnwell in South Carolina and at Hanford in Washington. Although each agreement state may establish its own inventory limits and administrative, surveillance, and reporting requirements, the same basic radiation protection standards apply as for Federally licensed facilities. Further, under provisions of the Clean Air Act Amendments of 1977, the states may set standards for radioactive emissions in the air which are more stringent than Federal standards.

EPA Waste Management Standards

The Environmental Protection Agency is responsible for developing standards applicable to all Federal radioactive waste management programs; these standards will be implemented in NRC regulations. EPA has published for public review the initial formulations of their standards.

In commenting on the draft of this Statement the EPA stated that they are presently proposing criteria and standards for radioactive waste management. These criteria and standards will be applicable to any disposal of high-level waste or spent nuclear fuel.

NRC Rules for Licensing of Geologic Repositories

The Nuclear Regulatory Commission has the statutory authority to license facilities used primarily for the receipt and storage of high-level radioactive wastes resulting from activities licensed under the Atomic Energy Act of 1954 and the Energy Reorganization Act of 1974. The Commission has indicated that regulations covering the licensing of Department of Energy disposal facilities will be issued as Part 60 of Chapter 10 of the Code of Federal Regulations (10 CFR 60). The procedural part of the NRC regulations was published for comment on December 6, 1979. It is expected that the technical portion of the regulations will be published for comment in late 1980.

DOT Regulations

Regulations governing the packaging, labeling, and shipping of radioactive materials, including radioactive wastes, are given in Title 49 of the Code of Federal Regulations and are too voluminous to be reproduced here. Included are descriptions of approved shipping containers for various quantities and types of radioactive materials, including performance criteria for protection against accidental damage. Limits on external levels of radiation are provided.

REFERENCES FOR SECTION 3.3

Title 10 Code of Federal Regulations. Part 20.

Title 40 Code of Federal Regulations. Part 120.

Federal Register. Vol. 25, p. 4402 et seq, May 18, 1960.

U.S. Energy Research and Development Administration. 1975. "Standards for Radiation Protection." In ERDA Manual of Operations, Chapter 0524. U.S. ERDA, Washington, D.C.

3.4 RISK AND RISK PERSPECTIVES

The potential environmental impact of nuclear waste isolation is often judged on the basis of a variety of risk and/or perceived risk issues. In this Statement, risk is defined as "probable loss." It is defined as the sum product of the magnitude of losses (the consequences) and the probability that these losses will occur. As defined, it does not discriminate between present or future events or between those of low probability/high magnitude and of high probability/lesser magnitude. Ordinary use of the term risk is not always consistent with this definition. For example, events of large magnitude, no matter how improbable, may be termed a large risk simply because of the size of the consequence. Similarly, when considerable uncertainty surrounds the estimate of probability or consequence, it might be said that a large risk is present. In both of these cases, the expected or most probable loss may be quite low.

Historically, society has tended to concentrate on minimizing the occurrence of high consequence events while giving little attention to low consequence events. An example is the required FAA safety certification of airplanes versus the relatively minor safety requirements for automobiles (seatbelts, safety glass, etc.). Americans are killed by the tens of thousands per year in auto accidents and by hundreds in airplanes. Yet it appears much more attention if not concern is given to 100 plane deaths than to 100 auto deaths. There is justification for placing attention on potential catastrophic events if such events could affect society's ability to recover from the catastrophic events. However, it is important to keep in mind that the amount of risk is not the only consideration in society's assessment of risk. Consideration of the benefit associated with that risk (or why the risk is being taken) also places the risk in perspective. The risk analyses in this Statement do not attempt to quantify the benefit associated with the generation of electricity which results in the production of nuclear waste.

This Statement considers the societal risk of the predisposal waste management technologies, the risk of operating a repository and the risk of long-term loss of containment or isolation. Two approaches to analyzing long-term risk are presented below: comparative hazard indices for both radioactive and non-radioactive materials including nuclear wastes, and the long-term analysis and risks associated with various scenarios for the release of radionuclides from deep geologic burial to the biosphere (consequence studies).

3.4.1 Hazard Indices

Hazard indices are based on estimates of potential risk of released radionuclides compared to other risks. The hazard indices can show whether the quantities of toxic radioactive waste exceed the toxic quantities of other chemicals and substances routinely handled in our society. A number of hazard indices have been developed which are useful in varying degrees in characterizing the risk. They are summarized in Appendix H of Volume 2. Hazard indices associated with radioactive materials are considered useful to the extent that the comparisons inform the reader about the magnitude of hazard compared to more familiar hazards.

One such hazard index is based on the amount of water required to bring the concentration of a substance to allowable drinking water standards. In the present case the amount of water required to bring the quantity of uranium ore (0.2% U_3O_8) necessary to make 1 MT of reactor fuel to drinking water standards (7×10^{-2} g/ℓ) was used as a basic hazard index. Assuming enrichment of ^{235}U to 3%, about 3,400 MT of ore would be required (95% recovery to make 1 MT of fuel. The hazard index of natural uranium of this quantity of ore is 8.7×10^7 m^3. The hazard index of the radionuclides in 1 MT of spent fuel was calculated based on 10 CFR 20 drinking water standards and summed for various times after the spent fuel was removed from the reactor. The hazard index for high-level waste from uranium-plutonium recycle was calculated in a similar way. Division by 8.7×10^7 m^3 made the hazard index relative to 0.2% uranium ore. In addition the hazard index of various ores was calculated relative to the volume of uranium ore equivalent to 1 MT of reactor fuel. These indices are presented in Table 3.4.1.

TABLE 3.4.1. The Relative Toxicity (Hazard) of Various Ores Compared to U Ore (0.2%)

Type of Ore	Average Ore	Rich Ore
Arsenic	1	10
Barium	5	20
Cadmium	28	120
Chromium	170	230
Lead	40	100
Mercury	460	3800
Silver	1	7
Selenium	70	220

The hazard index for spent fuel and high-level waste is shown in Figure 3.4.1, together with similarly developed hazard indices for ranges of common ores.

As seen in Figure 3.4.1 the hazard index for spent fuel or reprocessing waste from uranium-plutonium recycle relative to the ingestion toxicity of the volume of 0.2% uranium ore necessary to produce 1 MT of reactor fuel is on the order of that for rich mercury ores at about 1 year after removal of the spent fuel. The hazard index is on the order of that for average mercury ore at about 80 years. By 200 years the index is about the same as average lead ore. By 1500 years the relative hazard index for high-level waste is the same as the ore from which the fuel was made. For spent fuel the relative hazard index is about the same as the ore from which it came at about 10,000 years.

It is not suggested that spent fuel or high-level waste are not toxic. They are highly dangerous if carelessly introduced into the biosphere. It is, however, suggested that where concern for the toxicity of ore bodies is not great, then spent fuel or high-level waste should cause no greater concern particularly if placed within multiple-engineered barriers in geologic formations at least as, if not more, remote from the biosphere than these common ores.

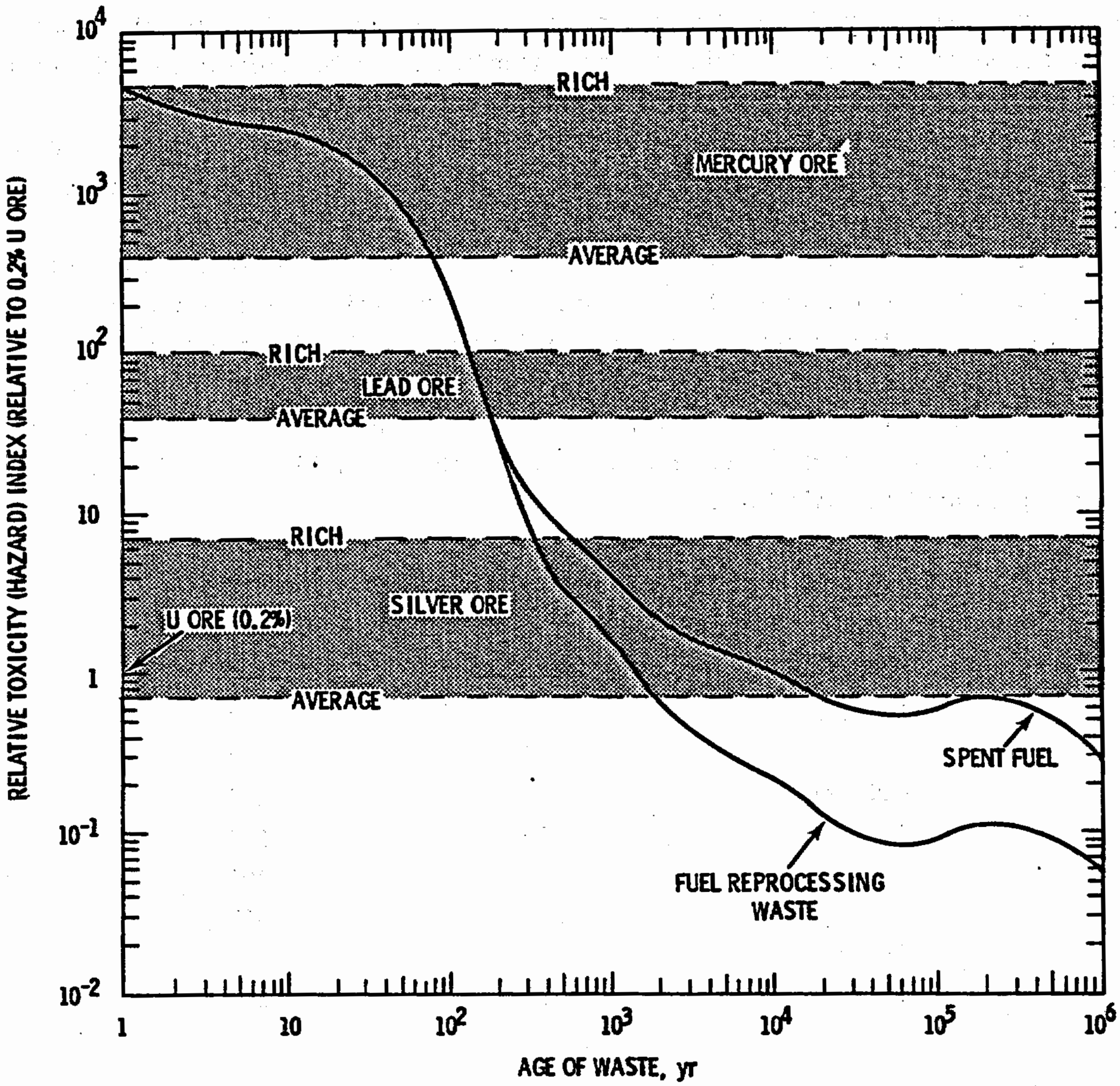

FIGURE 3.4.1 Toxicity of Spent Fuel and Reprocessing Waste from Uranium-Plutonium Recycle Relative to 0.2% Uranium Ore Necessary to Produce 1 MT of Reactor Fuel

Hazard indices generally neglect major confinement features such as the waste concentration (Hill 1977, Lash 1976), release mechanisms and dynamics (de Marsily 1977), and aspects of the food chain pathways. The hazard indices for the most part do not characterize the population exposures associated with conceivable natural and man-induced disruptive events--the key aspects of a risk assessment.

3.4.2 Consequence Analysis and Risk Assessment

Consequence analysis is the estimation of the effects of postulated accidental releases of radionuclides. Risk assessment is the calculation of the consequences of the spectra of possible accidental releases multiplied by their probabilities and summed to give a total risk. In this sense, the EIS does not present a complete risk assessment. The technique for such an assessment is still under development.

Since long-term repository containment cannot be demonstrated by short-term test, mathematical models must be relied on to predict the long-term behavior of the repository. Risk assessment is thus dependent on the development of reasonable predictions of the long-term behavior of the processes and phenomena that could occur within the repository system. The risk assessment under development for geologic isolation is taking the form described in the following methods.

3.4.2.1 Disruptive Events

Many geologic events and processes occur because of the long-term motion of the earth's plates with their associated stresses and strains, and by the action of long-term weather patterns associated with a variety of astrophysical and earth phenomena. Many of these phenomena are predictable (usually with an element of randomness); others can only be assigned an estimated site-dependent probability of occurrence. More specifically the key interest in predictive modeling is whether a site (selected by virtue of historical stability) will change to an unstable area (e.g., active faulting, volcanism, significant ground- and/or surface-water activity, etc.).

Potential disruptive phenomena that could affect a repository have been categorized as natural processes, natural events, man-caused events and repository-caused processes and are listed in Table 3.4.2.

The science of geology has tended to concentrate on predicting the location of ores and fossil fuels and to explain the structure of the earth. Nuclear waste isolation appears to be the first subject of large interest in long-term predictive geology. Many geologists have recently been engaged in the development of suitable predictive geologic models and/or scenarios. This research is concentrating on specific sites as well as global processes.

To be complete, risk assessment must include all significant sources of risk and must predict the condition of the repository and surrounding area following failure, the time of failure occurrence and its probability of occurrence. This evaluation is called "Scenario Analysis" (Burkholder 1978, Greenborg et al. 1978). In general, these evaluations employ models that are very complex and require the capabilities of electronic data processing. Confidence in the models can be increased by comparing the results of the models to natural systems which exist and adjusting the models until a reasonable degree of conformance is reached. This concept of calibration and verification has been employed in the hydrology models discussed below.

3.4.2.2 Lithosphere/Atmosphere Transport

This risk assessment process includes both lithospheric (by ground water) and atmospheric (by airborne and other surface processes) radionuclide transport analysis. The physicochemical processes governing ground-water movement and transport of pollutants are sufficiently understood that mathematical models can be formulated. However, these models require measured physicochemical parameters representing the specific site in order to simulate the system. These data are seldom adequate in terms of quantity and quality. However,

TABLE 3.4.2. Potential Disruptive Phenomena for Waste Isolation Repositories

Natural Processes	Natural Events	Man-Caused Events	Repository-Caused Processes
• Climatic Fluctuations	• Flood Erosion	Improper Design/Operation:	Thermal, Chemical Potential, Radiation, and Mechanical Force Gradients:
• Sea Level Fluctuations	• Seismically Induced Shaft Seal Failure	• Shaft Seal Failure	• Induced Local Fracturing
• Glaciation	• Meteorite	• Improper Waste Emplacement	• Chemical or Physical Changes in Local Geology
• River Erosion		Undetected Past Intrusion:	• Induced Ground-water Movement
• Sedimentation		• Undiscovered Boreholes or Mine Shafts	• Waste Container Movement
• Tectonic Forces		Inadvertent Future Intrusion:	• Increase in Internal Pressure
• Volcanic Extrusion		• Archeological Exhumation	• Shaft Seal Failure
• Igneous Intrusion		• Weapons Testing	
• Diapirism		• Nonnuclear Waste Disposal	
• Diagenesis		• Resource Mining (mineral, hydrocarbon, geothermal, salt)	
• New or Undetected Fault Rupture		• Storage of Hydrocarbons or Compressed Air	
• Hydraulic Fracturing		Intentional Intrusion:	
• Dissolution		• War	
• Aquifer Flux Variation		• Sabotage	
		• Waste Recovery	
		Perturbation of Ground-water System	
		• Irrigation	
		• Reservoirs	
		• Intentional Artificial Recharge	
		• Establishment of Population Center	

those data that can reasonably be obtained can be combined with a model to gain valuable insight. Some ground-water and transport models have been calibrated (Gupta and Pinder 1978, Kipp et al. 1976, Cole 1979) through adjustments of parameters to simulate measured behavior and thus can be used with some confidence in forecasting. These models have also been verified (Kipp et al. 1976, Ahlstrom 1977, Robertson 1977) by showing that they duplicate past trends in water table changes and contaminant transport in field situations.

Similarly airborne transport of ejected or reentrained radionuclide aerosols, subsequent uptake by biota, food chain pathways and exposure to and ingestion by man can be evaluated for specific sites.

REFERENCES FOR SECTION 3.4

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3.5 NONTECHNICAL ISSUES

Many of the issues concerning the management and disposal of radioactive waste do not confine themselves to strictly technical aspects of the problem. "Nontechnical issues" refers to broad social, political, and institutional concerns. This discussion is, in large part, based upon a Conference on Public Policy Issues in Nuclear Waste Management and on a recent report (Hebert et al. 1978).

The first part of this discussion organizes the nuclear waste issues into a smaller subset of issues and describes various positions on the issues. Further, the response to the issues raised by government agencies is discussed. The second part of this discussion examines in detail two areas of concern: short-term institutional arrangements and institutional arrangements for the long term.

3.5.1 Social Issues

A major issue concerning some people is the balancing of risks and benefits between this generation and future generations. One position on the issue is: at present transformation of the long-lived radioactive wastes into more short-lived forms is not feasible. As a result, future generations will have a burden of surveillance and monitoring, of risk to health and safety, and of corrective action should a containment breach occur, either from human or natural causes. Those holding this view state since this burden is difficult to specify and since the nation can afford to forego nuclear power benefits, production of more wastes would be morally irresponsible. An opposite position stresses that the risk exported to future generations is not unique to radioactive waste, is lower than commonly accepted risks, is a threat to relatively few people, and is low because of manmade and geologic barriers. Such low risk does not constitute an unfair burden given the benefits of nuclear power. A third position on this issue takes a more global view. Those with this view state that the issue of waste should be considered in the context of the benefits and costs and risks of all energy sources, not just nuclear power. For example, the problem of nuclear wastes should be viewed in the context of the benefit of preserving fossil fuels for future generations.

The issue of distribution of risk between generations is being examined by the Department of Energy and also by EPA and NRC. Early draft criteria by EPA have been explicitly concerned with this problem and reviewed in a public workshop held in Denver on March 30, 1980 (43 FR 2223). In his February 12, 1980 message on waste management, the President stated that his paramount objective is to "protect the health and safety of all Americans, both now and in the future." The Department of Energy in its Statement of Position on the Waste Confidence Rulemaking Hearings (DOE-NE-0007) takes recognition of this issue in its stated performance objectives, especially Objective 2, which specifies isolation for 10,000 years with no prediction of significant decrease in isolation thereafter, and Objective 5, which stresses conservatism in technical approach to provide assurance that regulatory standards can be met.

A second issue involves the need for candor. Concern has been expressed that information provided by the government and the nuclear industry concerning such events as the leaks at the Hanford, Washington, site has not been timely or relevant. However, since the mid-1950s there has been a large number of technical articles on nuclear power. Some take this as evidence of candor, while others see the flood of articles as an attempt to confuse the layman and increase reliance on the technical expert.

The President, in his February 12, 1980 message, noted that past governmental efforts to manage radioactive wastes have neither been technically adequate, nor have they sufficiently involved states, local governments and the public in policy and program decisions. The message established a program with mechanisms for full participation of these groups and continuous public review. The Department of Energy is fully committed to this program.

A third issue, public involvement, was a major topic at the Conference on Public Policy Issues (NSF 1976). Panelists at this conference generally agreed with the position that any person, group, or institution wanting to be involved in nuclear waste policy decisions has that right. Conference participants also pointed out that public participation does not guarantee sensible decisions nor an enhanced understanding of the issue. While general agreement was that final decisions should rest with the Federal government, some urged very strong public input on nuclear waste decisions via such mechanisms as state initiatives.

As stated above, the President's message has mandated full public participation in waste management policy decisions. Prior to this message, the Department of Energy held five public meetings in various regions of the country to seek public comment on the draft of this Statement in addition to the usual written comments. As a result of this input, this Final Statement has undergone extensive revision. Volume 3 of this Statement documents the extent of this revision. Further, the Interagency Review Group (IRG) received extensive public comment on their report dealing with nuclear waste management policy.

A fourth issue is that of uncertainty. Uncertainty pervades the technical and non-technical discussion about nuclear waste. The major uncertainties relating to nuclear waste involve: 1) effects of small doses of radiation received at low dose rates over a long time, 2) uncertainty about the ability to isolate nuclear wastes from the biosphere, and 3) uncertainty about human fallibility and malevolence. Some react to the uncertainty with caution and may urge a go-slow approach to waste isolation, while others feel that the uncertainties are sufficiently low to proceed with a waste isolation and disposal program.

In its Statement of Position for the "Waste Confidence" Rulemaking (DOE-NE-0007) the Department of Energy proposes a technically conservative approach to compensate for the perceived uncertainties in the ability to predict natural phenomena over long periods of time. The approach will utilize conservative design parameters, large margins for error, and multiple engineered and natural barriers in a step-by-step approach to implementation which will permit the capability of corrective action, should processes not operate as expected.

A fifth issue is that of equity. Some feel that those who live near a waste repository may be said to bear a greater risk in proportion to their benefit than do those remote from the repository. Some feel that those near the repository may not even benefit from the nuclear power which produced the waste. Another position stresses that people indirectly benefit from nuclear power because they buy products made with electricity from nuclear power and, therefore, such equity issues are less valid.

The Department of Energy is considering the feasibility of regional repositories, (i.e., repositories which serve the needs of the surrounding region) partly in response to concerns about equity (see discussion in Section 5.3). Under various scenarios there will be a need for more than one repository for a nuclear economy of 250 GWe by year 2000 (e.g., Case 3 in Section 3.2).

Concern about safeguards is a sixth issue. This concern hinges largely, though not exclusively, on the fact that plutonium, produced in the process of nuclear power production, is used in nuclear weaponry. Commercial fuel cycles which separate plutonium or other material with potential use in weapons raise the concern that they might be used for clandestine weapons development. Accounting for such material has been seen by some as inadequate. Some also worry that security against nuclear threats can only be achieved by intolerable infringements on personal freedom, while others feel that this is not the case. There is also a large difference in the perception of how difficult it is to build a bomb, ranging from the belief that one only needs access to a public library to a belief that it is a highly risky and technically challenging task requiring a sophisticated manufacturing capability.

The Department of Energy has an active research program for developing and improving safeguard and physical security methods that deal with transportation, storage and handling of radioactive materials. The NRC has promulgated and enforced safeguards and physical protection regulations for special nuclear materials such as plutonium (10 CFR 73).

Alternatives to nuclear power form a seventh issue area; that is, how one perceives conservation and other energy production alternatives affects perceptions of nuclear waste. The belief that cheaper, safer, less-polluting alternatives to nuclear power are available would incline the holder of that belief to oppose the production of nuclear wastes. Some, however, feel that nuclear power is superior to currently available technologies and therefore are willing to accept the radioactive waste problem. Even if no further nuclear weapons production or power generation occurred, an inventory of wastes from past activities would need to be stored or disposed.

In its Statement of Position at the "Waste Confidence" Rulemaking, the Department of Energy proposed in Objective 7 that disposal concepts selected for implementation should be independent of the size of the nuclear industry (DOE/NE-0007). This is in accord with the President's statement of February 12, 1980, which requires that waste disposal efforts proceed regardless of future developments in the nuclear industry. This EIS examines 5 cases of nuclear development ranging from termination of nuclear power in 1980 to full development to properly assess nuclear waste management systems (see Chapter 7).

An eighth issue area is the transportation of nuclear waste material. Concerns about accidents, sabotage, and thefts of material in transit are at the core of these concerns and so relate to the issues previously mentioned.

The U.S. Department of Transportation (DOT) is currently in a rulemaking process concerning transportation of high-level nuclear wastes (45 FR 7140). Further, current regulations of both DOT and NRC are considered to adequately protect public health and safety (49 CFR, Parts 173 and 177).

The irreversibility of geologic waste disposal is the core of a ninth issue. The argument has been made that because of its apparent irreversibility we should delay implementing geologic isolation until we are more certain that the wastes will not be used now or in the future. Other arguments for delay include keeping the wastes retrievable for 20 to 30 years in case something goes wrong in the repository or in case a better method is devised in this period. However, the argument has also been made that disposal methods that are technically impossible to reverse offer the best solution to isolating the wastes from man.

In its Statement of Position (DOE-NE-0007) of April 1980, the Department outlined its "step-wise" approach. This conservative approach would store a limited quantity of material under well understood conditions and then proceed in a series of small steps so that the material could be retrieved should unanticipated problems make the system unacceptable. NRC has also reflected this approach in a recently issued draft of possible technical regulations which would require the capability of retrievability for 50 years after emplacement operations have ceased. The ability to retrieve the wastes during the initial periods of operation is seen as one of the main advantages of mined geologic repositories.

The tenth issue area involves the distinction drawn between commercial and military wastes. Some have argued that no distinction should be made on the constraints of the management of the two wastes, while others have argued that they should be kept distinct because of the very different physical nature of the wastes.

The Presidential message of February 12, 1980 specifically directs that the radioactive waste management program seek to isolate and dispose of wastes from both civilian and military activities.

International responsibilities form an eleventh area of concern. The waste issue is larger than U.S. boundaries because of technology export and import of wastes and because of possible international solutions to the waste problem. Worldwide releases of radioactivity may cause health and genetic problems which respect no national boundaries. Further, concern has been expressed that in lesser developed countries, cost concerns could lead to an inadequate waste management plan. Since much of the nuclear waste is now produced in foreign reactors, some of which are U.S. exports, the argument has been made that the U.S. must show leadership in solving the nuclear waste problem. An international waste management authority has been proposed to handle these problems.

The Department of Energy is mindful of international responsibilities for nuclear waste and is participating in a number of bilateral and multilateral programs to deal with nuclear waste. Examples are a cooperative investigation with Sweden at a mine in Stripa, Sweden, a cooperative agreement with the Federal Republic of Germany for exchange of technical information on waste disposal, and active participation in the International Atomic Energy Agency (IAEA).

A twelfth issue area is that of cost of waste management. Participants in the Conference on Public Policy Issues on Nuclear Waste Management showed general agreement that we must be willing to pay for an adequate disposal system. Some fear that adequate charges will not be assessed to provide perpetual care. Current regulations require a fee to be paid to the government at the time of transfer of the waste to Federal custody, although the size of this fee has not been determined.

The President's message of February 12, 1980 specified that "all cost of storage, including cost of locating, constructing and operating permanent geologic repositories will be recovered through fees paid by utilities and other users of the services and will ultimately be borne by those who benefit from the activities generating the wastes."

A final issue area, discussed more fully below, concerns institutions for controlling and managing nuclear waste. These concerns relate both to the short term, i.e., the period of time up to the closure of a waste repository, and to the long term, i.e., the period following closure for the hundreds of years during which the potential hazards of the waste remain. Some individuals contend that past mishaps and leaks involving military wastes are a basis for regarding the current institutional arrangements as inadequate. Others judge that current institutions have done an adequate job or that new arrangements will lead to better waste handling. Further the ability of institutions to monitor disposed waste in the long term is a key part of the issue area. Some feel that technical considerations will make such long-term monitoring unnecessary, while others feel that the waste has to be monitored for as long as 200,000 years and would be a formidable task. A more intermediate view is that monitoring might be required for several hundred years.

In the Department of Energy's Statement of Position for the NRC "Waste Confidence" Rulemaking (DOE/NE-0007), a proposed objective of the program was to provide reasonable assurance that wastes will be isolated from the environment for at least 10,000 years with no prediction of significant decrease in isolation beyond that time. Further governmental concern for this issue is shown by the proposed EPA criterion that a waste disposal system cannot rely on human institutions for a period of more than 100 years (42 FR 53262).

3.5.2 Institutional Issues

The following two sections briefly expand on short-term and long-term institutional concerns. These two sections discuss institutional concerns without reference to scale of the waste management system. Some have argued that institutional issues may potentially become much more severe with increasing scale (LaPorte 1978).

3.5.2.1 Short-Term Concerns and Institutional Design

Technical solutions to waste management problems are not self-implementing. They require institutions, either those existing or ones yet to be created, to make them work.

Setting up a waste management program therefore requires institutional choices: whether to rely on existing organizational arrangements or to develop new ones. This section discusses some considerations regarding choice of one or another set of organizational arrangements for waste management. Additionally, the institutions discussed below should function in conjunction with the engineered design as part of the overall waste management system.

The Department of Energy (DOE) is currently responsible for establishing programs leading toward the treatment, storage, and disposal of nuclear wastes. The Environmental Protection Agency is responsible for setting generally applicable environmental standards for radioactive waste (3 CFR). The Nuclear Regulatory Commission is responsible for implementing these standards, establishing regulations and policies, and licensing commercial waste management facilities (10 CFR 20 301, 42 U.S.C. 5842). State governments (in agreement states) license and regulate low-level burial sites (42 U.S.C. 2021). The Department of Transportation (DOT) shares responsibility for regulation of the transportation of wastes with NRC (38 F.R. 8466, March 22, 1973).

A number of organizational options are available for the management and disposal of nuclear waste. Below are listed four such options: 1) Federal agency; 2) government corporation; 3) government-owned, contractor-operated facility; and 4) contractor-owned, contractor-operated facility. In a Federal agency, waste management functions would be performed directly by Federal agency employees who are ordinarily members of the Federal civil service. A government corporation is a Federally chartered organization with its own legal personality distinct from that of the Federal government. It is exempt from civil service rules, thus allowing the managers of the corporation to retain control over all aspects of personnel management. A government-owned, contractor-operated arrangement is similar to the government corporation, especially in the private contractor's flexibility with respect to personnel practices and financial systems. A contractor-owned, contractor-operated arrangement differs chiefly in that the contractor's financial commitment is much heavier than under a government-owned, contractor-operated arrangement.

In addition to consideration of organizational options, a knowledge of the basic regulatory functions is useful in assessing the adequacy of institutional arrangements for managing and disposal of nuclear waste. The function of regulating the commercial nuclear waste management system includes the tasks of standard-setting, licensing, technical review, inspection, and enforcement. Below is a brief discussion of each task.

Standard-setting and licensing are often done by the same organization. Sometimes, however, one agency (such as EPA) has the task of setting general rules for how tasks must be done (performance standards), while another agency (such as NRC) has the task of applying those general standards to a specific case, and of granting a license to operate when proper conditions have been met.

A technical review of a proposed action for its scientific adequacy may increase the safety of the waste management system by helping to avoid errors at key decision points. Reviewer independence is a valuable attribute; it reduces the opportunities for bias and, hence, the chances that a review will become automatic approval.

Inspection, the regular checking of the actual waste management operation to ensure that it is being performed in the proper manner, is one of the most critical functions in the entire waste management system. If other parts of the system break down, a good inspection system will detect them. If the inspection system itself fails, no one will know whether or not the waste management system is reliable.

The character of the enforcement function depends on whether private or public organizations are the target. In the case of private organizations, credible penalties, such as fines and license revocation, are available. But these sanctions cannot be expected to have the same effect on public organizations, which are less influenced by economic incentives.

3.5.2.2 Institutions in Long-Term Nuclear Waste Management

A number of concerns have been raised regarding the role that human institutions may have in the long-term management of nuclear wastes. Controversy exists concerning: 1) the need for any human institutions to be involved in long-term management, and 2) whether human institutions could actually carry out any functions that might be required of them over the long term.

These discussions are speculative. Historical examples of the behavior and durability of human institutions are the only data that can be applied to the speculations about the potential future stability and performance of institutions. However, to predict what the world will be like 50 to 100 years from now, let alone in several centuries, is very difficult.

Human institutions might enhance safety by accurately predicting the occurrence of the natural events which could compromise the repository (e.g., earthquakes, floods), and in responding to them to reduce consequences. Control over these massive events is not likely.

Human actions that might produce a release of radioactive material from a repository have been grouped into three categories: 1) major catastrophic events, such as nuclear war, 2) direct action against the repository, such as sabotage, drilling and exploration, and excavation, and 3) lapses in monitoring, such as being unaware of a breach in the containment.

Three sets of factors appear pertinent in assessing the institutional role in long-term waste management: 1) the functions that can or should be performed by the institutions, 2) the subjective need for these functions, and 3) the likelihood that the functions will be performed at any given point in time.

Three general categories of functions might increase the safety of a waste repository and mitigate the consequences of potential accidents:

1. Control and management--including monitoring of security and physical integrity, performance of routine physical plant maintenance, and maintenance of a staff of people qualified to carry out technical tasks at the disposal site.

2. Monitoring--including observation of seismic, thermal, and radiological conditions to detect any releases or significant changes in site integrity.

3. Information transfer--including maintenance of records and data about the repository and its contents. Such information would be needed to effect repair of a site, to warn future generations about the dangers of the wastes, to inform people about the resource value of the contents, and to prevent an intrusion into the repository at some time in the distant future.

It has been suggested that human institutions could provide an increment of safety if monitoring, surveillance, and security operations are carried out during the first few centuries after a repository is closed. Human activites would provide a backup to the engineered system. This backup system would have the function of predicting the occurrence of natural hazards, preventing human intrusions, and responding to any anomalies that occurred at repository sites. These last two functions were seen by some to be especially significant in the mitigation of repository accidents.

Predictions are very difficult to make with certainty about whether future societies would find the task worthwhile to support institutions to carry out the functions noted above. It has been argued that it is up to future generations to decide for themselves whether to carry out these functions. Predictions are also impossible to make on whether information can be conveyed across millenia, or whether organizations can be established that could last for such time periods. The focus of assessment has been to analyze any evidence to suggest that if organizational and institutional continuity were necessary, could institutions be established in the present that might survive long enough to carry out their tasks?

The analysis of these issues is, of necessity, purely speculative, and based on historical examples that provide no firm basis for making predictions. However, some examples suggest that complex information in abstract form can be maintained over thousands of years.[a] The sacred books of major religions and the hieroglyphics of ancient Egypt are examples. Furthermore, many functional organizations, such as the U.S. Government, have survived for a century or more while carrying out roughly the same tasks. A few, such as the British political system, have survived for nearly a millenium. Of course, how much information has been lost in historical times is not known.

The principal conclusions of this analysis are:

- There are no reasons in principle to indicate that human institutional functions cannot survive for hundreds of years, given reasonably stable political systems. However, no strong evidence exists that such functions will, in fact, survive.
- Technical information can be maintained for a very long time if a culture remains literate and the information has a continuing utilitarian value.
- Waste management systems adopted in the present time period should place minimal, if any, reliance on any human management after the repository is closed.

(a) Additionally, no prior known civilization has had both the mass education and communication systems that presently exist.

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CHAPTER 4

PREDISPOSAL SYSTEMS

After radioactive wastes are generated and before their disposal, several predisposal operations are required. The combination of these operations is referred to in this Statement as the predisposal system. The system operations include treatment and packaging to prepare the waste for the specific requirements of a disposal option, interim storage if the treated waste cannot be shipped immediately to a disposal site, and shipment to interim storage and/or to a disposal site. Decommissioning of the waste management facilities, although not a predisposal operation, is discussed in this chapter because it produces wastes which must be managed in a manner similar to those wastes produced by fuel reprocessing and MOX fuel fabrication plants.

This chapter provides examples of processes and facilities that could be used to carry out these predisposal operations for both the once-through cycle and the reprocessing cycle. The processes and facilities described here are not dependent to a significant degree on the size of the nuclear system served. For each required step, one or more concepts have been examined in detail to characterize the environmental impacts of construction, operation and decommissioning, the impacts of potential accidents, the dollar cost of construction and operation, and the safeguard requirements. Summary results of these evaluations are presented here. Detailed results are available in DOE/ET-0028 and DOE/ET-0029.

All of the concepts evaluated here are considered to represent available technology; that is, enough information is available to initiate design and construction of full-scale facilities, although varying degrees of design verification testing may be required. Brief descriptions are also provided of a number of alternative high-level waste treatment concepts that do not represent available technology but have attractive attributes that make them potential alternatives.

4.1 RELATIONSHIP OF PREDISPOSAL OPERATIONS TO DISPOSAL AND PROGRAM ALTERNATIVES

The relationships of the predisposal operations to the unique system requirements for each disposal alternative, for both the once-through and the fuel reprocessing cycles, are described in this section. The individual components of the predisposal systems are then described and analyzed in subsequent sections.

4.1.1 Predisposal System for the Once-Through Cycle

A simplified diagram of the predisposal waste management system for spent fuel in the once-through fuel cycle is shown in Figure 4.1.1. For the example predisposal system assumed here, the spent fuel is stored at the reactor storage basins for a minimum of 5 years. The fuel may be stored there for a longer period if a disposal facility is not available and if capacity is available at the reactor. The fuel is then shipped to a

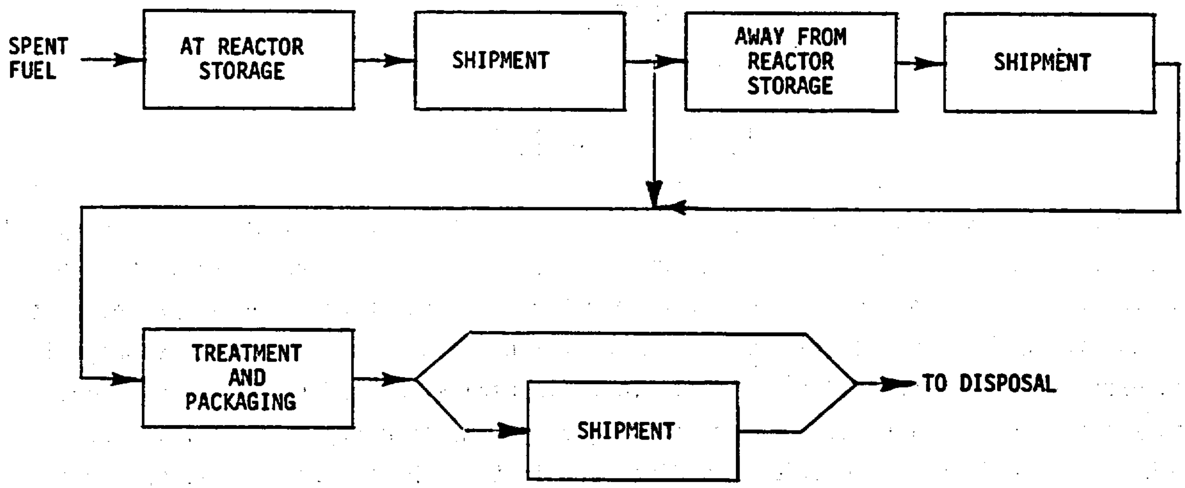


FIGURE 4.1.1 Predisposal Waste Management System for Spent Fuel in the Once-Through Fuel Cycle

treatment and packaging facility if a disposal facility is available. If a disposal facility is not available, the fuel is assumed to be shipped to an away-from-reactor (AFR) storage facility.[a] When a disposal facility is available, the fuel is shipped there for treatment and packaging prior to disposal. Alternative approaches include having packaging facilities located separately from disposal facilities with extended storage of packaged fuel before disposal.

The types of operations and facilities considered in this Statement for each of the disposal alternatives are identified in Table 4.1.1. This table shows that the initial storage and shipment operations are identical for all of the disposal alternatives. The differences in the predisposal systems are in the treatment and packaging and final shipment to disposal. Four of the eight alternatives to mined geologic disposal can utilize the same treatment and packaging options as mined geologic disposal; however, three of these require ocean ship transport to the final disposal site. Four of the alternatives can only be utilized in the once-through cycle if the spent fuel is first dissolved as in a reprocessing cycle. Two of these alternatives require disposal as liquid high-level waste. In these two cases, no shipment to disposal is required because the treatment facility and the disposal facility are located on a common site. The transmutation alternative requires, in addition to dissolution of the fuel, complex chemical partitioning, target fabrication, and irradiation. Space disposal requires, in addition to dissolution of the spent fuel, a process to convert the liquid waste into an encapsulated solid material. All of the alternatives that utilize a dissolution process would also generate considerable quantities of miscellaneous TRU waste. These would require the same treatment and handling as the comparable wastes produced in the reprocessing cycle described in the next subsection.

(a) AFR storage facilities were referred to as independent spent fuel storage facilities (ISFSFs) in DOE/ET-0028 and DOE/ET-0029.

TABLE 4.1.1. Predisposal Operations and Alternatives for Once-Through Cycle Disposal Options

Disposal Option	Shipment to Interim Storage	Interim Storage	Shipment to Treatment	Treatment and Packaging	Shipment to Disposal
Mined geologic	Rail and Truck	Water basin Alternatives include packaged fuel storage in: • Dry wells • Air cooled vaults • Surface casks	Rail and truck	Encapsulate individual assemblies Alternatives include: • Encapsulate multiple assemblies • Disassemble and encapsulate • Chop, voloxidize and encapsulate • Dissolve and convert to glass[a]	None if onsite or rail if offsite
Very deep holes	Same as above	Same as above	Same as above	Same as above	Same as above
Rock melting	Same as above	Same as above	Same as above	Dissolve and dispose as liquid[a,b]	Onsite disposal
Island	Same as above	Same as above	Same as above	Same as mined geologic island transports	Rail, ocean ship and island transporter
Subseabed	Same as above	Same as above	Same as above	Same as mined geologic	Rail and ocean ship
Ice sheet	Same as above	Same as above	Same as above	Same as mined geologic	Rail, ocean ship and over-ice vehicle
Well injection	Same as above	Same as above	Same as above	Dissolve and dispose as liquid[a,b]	Onsite disposal
Transmutation	Same as above	Same as above	Same as above	Dissolve, partition, fabricate targets, irradiate and reprocess targets[a]	Truck or rail to and from irradiation
Injection into Space	Same as above	Same as above	Same as above	Dissolve and convert to "cermet" matrix in capsules[a]	Rail to launch site; launch to orbit, see Section 6.1.8

(a) Spent fuel treatment involving dissolution produces TRU wastes requiring all of the TRU waste predisposal operations shown in Table 4.1.3. for reprocessing cycle wastes. These TRU wastes then probably will require mined geologic disposal.
(b) Disposal of spent fuel as an aqueous liquid in the rock melting and well injection options may not be feasible because of criticality questions.

4.1.2 Predisposal System for the Reprocessing Cycle

A simplified diagram of the predisposal waste management system for the reprocessing cycle is shown in Figure 4.1.2.[a] In this cycle, wastes requiring disposal are produced at the fuel reprocessing plant (FRP) and at the mixed-oxide fuel fabrication plant (MOX-FFP). These wastes are assumed to be treated and packaged at the site where they are produced, either the FRP or MOX-FFP. They are then shipped to interim storage if a disposal facility is not available; finally, they are shipped to a disposal facility.

The operations and facilities required for the predisposal system for management of the high-level waste are shown in Table 4.1.2. As in the case of spent fuel, four of the alternatives to mined geologic disposal can utilize the same treatment and interim storage processes as the mined geologic disposal option. Three of the alternatives, however, require ocean transport to the final disposal site. In the two cases where high-level waste is disposed of as a liquid, the only predisposal system facilities required for high-level waste are the interim storage facilities consisting of double-walled below-grade tanks. For the transmutation alternative, interim storage is assumed to be required for the liquid high-level waste in double-walled below-grade tanks prior to the partitioning processing. This storage requirement and the target recycle requirements are thus exceptions to the sequence of operations shown in Figure 4.1.2. For space disposal, as in the once-through cycle, the high-level waste solution is converted to a solid "cermet" matrix contained in special spherical capsules. Interim storage would be similar to that of spent fuel, but because of the shape of the container, it would have its own unique design requirement.

Various TRU waste materials must also be disposed of in all of the disposal concepts. Although it may be possible to dispose of some of these materials after treatment in the same facility used for disposal of the high-level waste, it is assumed here that these materials are always sent to a mined geologic repository regardless of the disposal option selected for high-level waste. The operations and facilities considered for the predisposal system for these waste materials are shown in Table 4.1.3.

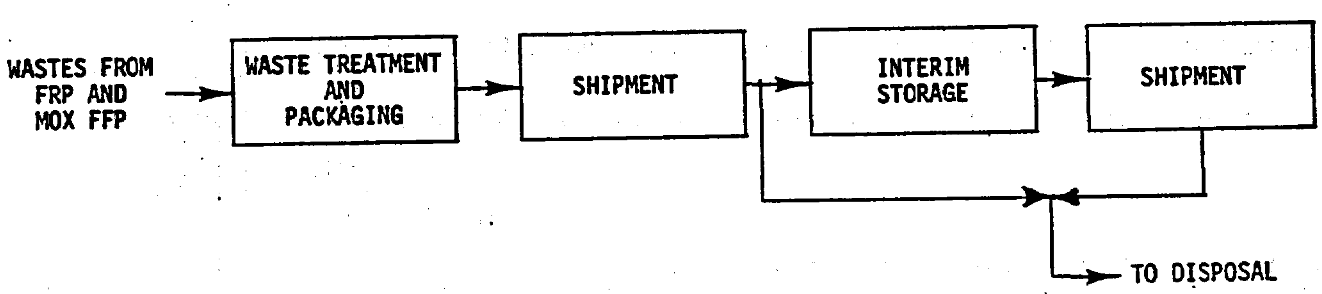


FIGURE 4.1.2. Predisposal Waste Management System for Fuel Reprocessing Plant and MOX-Fuel Fabrication Plant Wastes in the Fuel Reprocessing Cycle

(a) For a description of the fuel cycle prior to waste generation at the FRP and the MOX-FFP, see Figure 3.2.2.

TABLE 4.1.2. Predisposal Operations and Alternatives for Reprocessing-Cycle High-Level Liquid Wastes

Disposal Option	Waste Treatment	Shipments to Interim Storage[a]	Interim Storage	Shipment to Disposal
Mined geologic	Convert to stable solid such as a calcine, a glass,[b] a synthetic mineral, a metal matrix, etc.	Rail[b] or truck	Water basins and/or air-cooled sealed casks[b]	Rail[b] or truck
Very deep holes	Same as above	Same as above	Same as above	Same as above
Rock melting	Not required	Not required	Double-walled tanks	Onsite disposal
Island	Same as mined geologic	Same as mined geologic	Same as mined geologic	Rail and ocean ship
Subseabed	Same as mined geologic	Same as mined geologic	Same as mined geologic	Rail and ocean ship
Ice sheet	Same as mined geologic	Same as mined geologic	Same as mined geologic	Rail and ocean ship
Well injection	Not required	Not required	Double walled tanks	Onsite disposal
Transmutation	Partition, fabricate targets, irradiate and reprocess targets	Not required	Double walled tanks	Truck or rail to and from irradiation
Injection into space	Convert to a "cermet" matrix in capsules	Same as mined geologic	Similar to mined geologic	Rail or truck to launch site; launch to orbit see Section 6.1.8

(a) A 5-year storage period in water basin facilities at the reprocessing plant is assumed before shipment to other interim storage.
(b) The example method of this Statement.

TABLE 4.1.3. Example Predisposal Operations and Alternatives Evaluated for Reprocessing-Cycle TRU Wastes for All Disposal Concepts

Non-High-Level Waste Type	Waste Treatment	Shipments to Interim Storage	Interim Storage	Shipments To Disposal
Fuel Residue[a]	Package in canisters without compaction.[b] Alternatives include: • Mechanical compaction of hulls • Hulls melting	In casks by rail[b] or truck	Dry-well facility[b] or concrete vault	In casks by rail[b] or truck
Failed equipment and other noncombustible waste	Failed equipment decontaminated and disassembled as required. Noncombustible waste packaged without treatment. Packaged in canisters, drums and boxes	Canisters in casks by rail[b] or truck. High dose-rate drums in casks by rail or truck.[b] Other drums and boxes in shielded overpacks or special containers by rail or truck[b]	Canisters in dry-well facility[b] or concrete vaults. High dose-rate drums in dry-well facility or concrete vaults.[b] Low dose-rate containers in unshielded buildings or outdoors with earth cover[b]	Same as to interim storage
Combustible waste	Incinerate and immobilize ash in cement[b] or bitumen. Alternatives include packaging without treatment	Drums in casks or shielded over packs or special containers by rail or truck[b]	High dose-rate drums in dry-well facility or concrete vaults.[b] Low dose-rate containers in unshielded buildings or outdoors with earth cover[b]	Same as to interim storage
Wet wastes and particulates	Immobilize in cement[b] or bitumen	Same as above	Same as above	Same as above
Gaseous and airborne wastes	Use high efficiency filters and process to remove I, C and Kr. Alternatives include ^3H removal	Recovered solids as above. ^{85}Kr not shipped	^{85}Kr stored on-site in special facility for pressurized gas cylinders. Other materials as above	Recovered solids as above. ^{85}Kr not shipped offsite

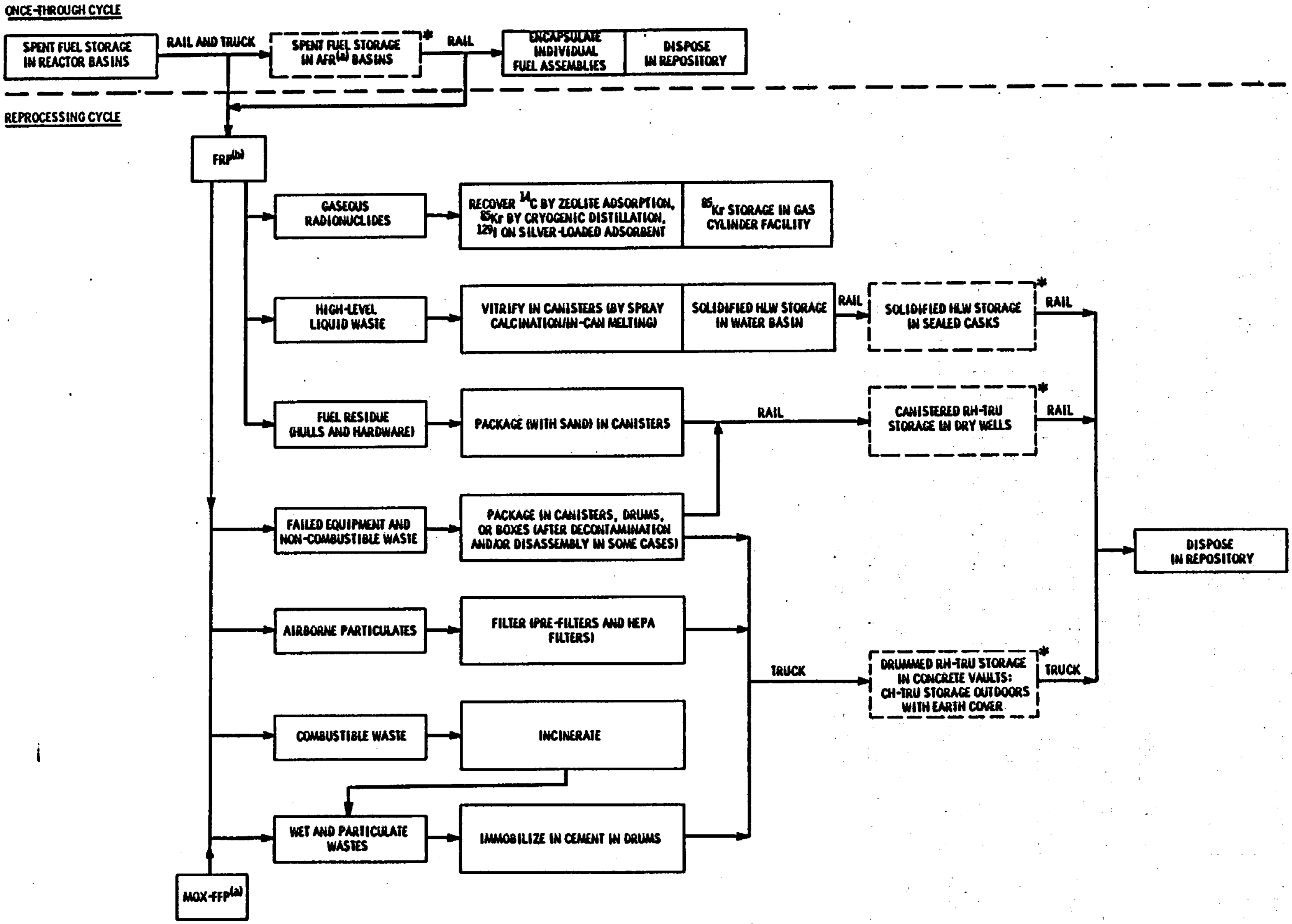
(a) Spent fuel cladding hulls and hardware that remain after fuel components have been leached out.
(b) The example method of this Statement.

Although they are not necessarily waste management functions, the spent fuel handling and storage operations that occur before reprocessing are, to be conservative, also included in the predisposal system in the system simulation analyses in Chapter 7. This includes the operations shown in Figure 4.1.1 prior to treatment and packaging.

4.1.3 Predisposal System Relationships to Program Alternatives

The predisposal systems for the preferred alternative, that is, a program leading to utilization of mined geological repositories, are listed with the mined geologic disposal option in Tables 4.1.1, 4.1.2 and 4.1.3. If the alternative program to develop several disposal options in parallel were to be adapted, some of the other predisposal operations shown in these tables might be utilized. For the no-action alternative, spent fuel would be stored indefinitely without either reprocessing or final disposal.

The predisposal waste management operations for the preferred alternatives are given schematically in more detail for both fuel cycles in Figure 4.1.3. These operations are discussed in more detail in Sections 4.3 to 4.6.



* FUNCTIONS IN DASHED BOXES DEPEND ON REPOSITORY START-UP DATE (AND/OR REPROCESSING START-UP DATE, IN THAT CYCLE)

(a) EXAMPLE DECOMMISSIONING MODE IS IMMEDIATE DISMANTLEMENT

(b) EXAMPLE DECOMMISSIONING MODE IS 30-YR PASSIVE SAFE STORAGE BEFORE DISMANTLEMENT

FIGURE 4.1.3. Example Predisposal Waste Management Operations for the Mined Geologic Disposal Option

4.2 UNTREATED WASTE CHARACTERIZATION

The quantities and composition of the wastes generated at each step in the post-fission LWR fuel cycle have been studied in detail. Quantities used in this Statement are based upon actual practice for processes that have been demonstrated and upon technical judgments for processes that have not yet been commercially demonstrated. The untreated initial wastes, termed primary wastes, are identified, described, and classified as the first step in defining the environmental impact of radioactive waste treatment. Additional details are presented in DOE/ET-0028 (Section 3.3).

The primary wastes are processed to form treated wastes suitable for disposal. It is anticipated that essentially all commercial wastes (on a Curie basis) or a large fraction (on a volume basis) will receive treatment. Treated wastes are of two types: 1) gaseous wastes that have been treated to reduce their activity levels so they can be released to the environment without harm to man, and 2) wastes that have been converted to a stable form suitable for disposal so that their radioactivity will remain confined and out of contact with man's environment.

Secondary wastes are generated in the treatment of primary wastes and in the subsequent handling of treated wastes. Secondary wastes are generated not only from initial waste processing, but also from the storage, transportation, and isolation steps. In most cases, the amount of secondary wastes is small in comparison to the amount of primary wastes; nevertheless, an assessment of the environmental impacts is not complete without including the effects of the secondary wastes. Treated secondary wastes are included with the treated primary wastes in Section 4.3.7.

Decommissioning wastes result from the operations employed to decommission retired nuclear fuel cycle facilities. These wastes must also be included in a complete analysis of the impacts of nuclear waste treatment; characterization of such wastes is presented in Section 4.6.

Many methods of classifying radioactive wastes are in use, based on the kind of radioactivity contained, the amount of radioactivity contained, the untreated physical form, the treated physical form, etc. In this Statement, wastes have been classified into categories based on their treatment requirement; i.e., all wastes requiring a similar treatment are included in the same category. The categories and a brief generic description of each are given in Table 4.2.1. The first three waste categories are specific to certain fuel cycles. Spent fuel as a waste is specific only to the once-through cycle, and high-level liquid waste and fuel residue are specific only to the reprocessing cycle. The last four waste categories listed in Table 4.2.1 are generated in almost every facility in which radioactive materials are processed, treated, or handled. Thus, both primary and secondary wastes of these categories are found throughout the LWR fuel cycles.

Radioactive wastes are also generally classified according to their content of transuranic (TRU) radionuclides (i.e., radionuclides with atomic number greater than 92). Because of the long half-lives and high radiotoxicity of some TRU nuclides, TRU wastes are

TABLE 4.2.1. Classification of Primary Wastes from the Post-Fission LWR Fuel Cycle

Waste Category	General Description
Spent fuel	Irradiated PWR and BWR fuel assemblies containing fission products and actinides in ceramic UO_2 pellets sealed in Zircaloy tubes. Intense radioactivity.
High-level liquid waste	Contains about 0.5% of the U and Pu in the spent fuel and over 99% of the fission products and other actinides. Intense radioactivity.
Fuel residue	Includes short segments of Zircaloy tubing (hulls) remaining after UO_2 is dissolved and stainless steel assembly hardware.
Gaseous	Predominately two types: 1) large volumes of ventilation air, potentially containing particulate activity, and 2) smaller volumes of vessel vent and process off-gas, potentially containing volatile radioisotopes in addition to particulate activity.
Compactable and combustible wastes	Miscellaneous wastes including paper, cloth, plastic, rubber, and filters. Wide range of radiation levels dependent on source of waste.
Concentrated liquids, wet wastes, and particulate solids.	Miscellaneous wastes including evaporator bottoms, filter sludges, resins, etc. Wide range of radioactivity levels dependent on source of waste.
Failed equipment and noncombustible wastes	Miscellaneous metal or glass wastes including massive process vessels. Wide range of radioactivity levels dependent on source of waste.

considered more hazardous than non-TRU wastes. Present regulations governing disposal of TRU wastes are more stringent than those governing disposal of non-TRU wastes. Non-TRU wastes are eligible for disposal by surface burial and, except for gaseous and airborne wastes, some of which contain non-TRU radionuclides of special concern (^{129}I, ^{85}Kr and ^{14}C), management of these wastes is outside the scope of this Statement. However, data on the characteristics of untreated post-fission non-TRU wastes are included in DOE/ET-0028 (Section 3.3) along with those of the TRU wastes.

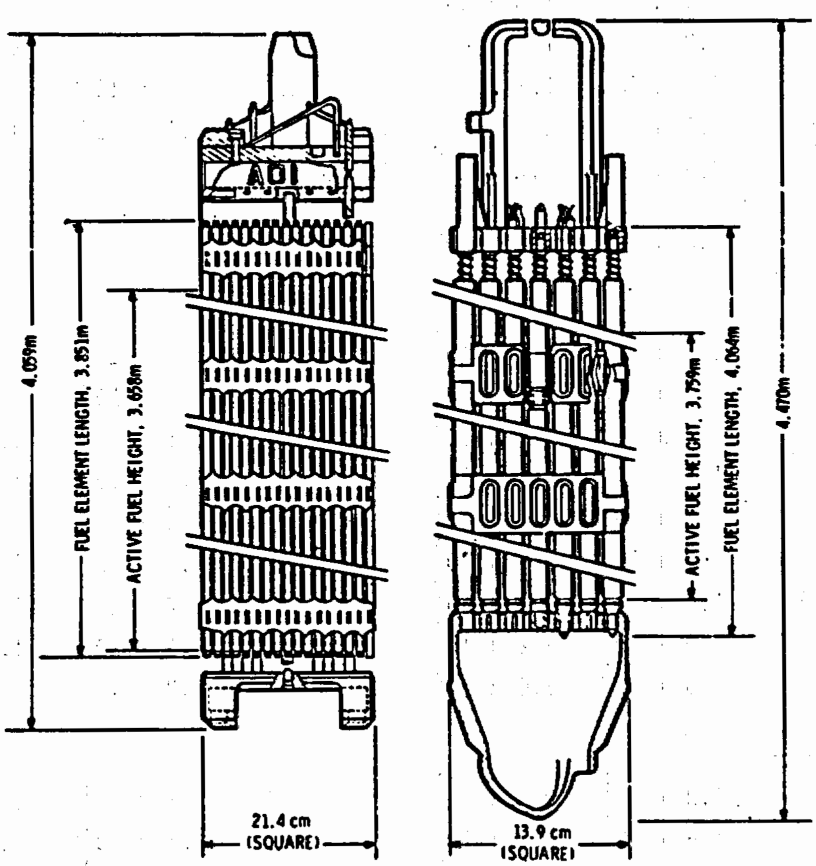
In current practice, a TRU waste is considered to be one that contains more than 10 nanocuries of transuranic alpha activity per gram of waste. However, spent fuel as waste and high-level waste that results from processing spent fuel, which contain high levels of transuranic activity, are considered as a separate high-level waste category. Raising the dividing line between TRU and non-TRU wastes from 10 nCi/g to 100 nCi/g has been proposed by EPA. Because these low concentrations are often difficult to measure in wastes, we assume in this Statement that all wastes from locations that might cause contamination levels above 10 nCi/g of waste are considered to be TRU-suspect and are combined with known TRU wastes for treatment.

In order to relate waste quantities to electric energy generation and to facilitate comparisons between alternative nuclear fuel cycles, the waste volumes and activities in this section are given per GWe-yr. One GWe-yr (or 8.8×10^9 kWh) is equivalent to the annual output of one of the largest nuclear power plants operating today (a 1250 MWe plant operating for one year at 80% capacity produces 1 GWe-yr of electricity). One GWe-yr also corresponds to the annual electrical energy consumption of about one million people in the U.S. (The total electric utility sales in 1978 amounted to about 230 GWe-yr.) For the generic LWR fuel cycle upon which this Statement is based, 38 MT of UO_2 or mixed UO_2-PuO_2 (MOX) fuel must pass through the cycle to generate 1 GWe-yr.

4.2.1 Once Through-Cycle Wastes

The only primary waste in the once-through fuel cycle within the scope of this Statement is the spent fuel itself. Two basic types of LWR fuel are in use today: pressurized water reactor (PWR) fuel and boiling water reactor (BWR) fuel. The reference PWR and BWR fuel assemblies defined for this generic Statement are described in Figure 4.2.1. Fuel for specific plants may vary somewhat from these descriptions.

For the purpose of describing radioactivity content of the wastes here, an example fuel composition based on a representative mixture of PWR and BWR fuel assemblies was developed. However, the system simulation results presented in Chapter 7 are based on explicit PWR and



	PWR	BWR
FUEL ELEMENT ARRAY	17 x 17	8 x 8
ASSEMBLY TOTAL WEIGHT, kg	658	283.9
URANIUM/ASSEMBLY, kg	461.4	188.7
MO_2/ASSEMBLY, kg	523.4	214.1
ZIRCALOY/ASSEMBLY, kg	108.4	59.6
HARDWARE/ASSEMBLY, kg	26.2	10.2
TOTAL METAL/ASSEMBLY, kg	134.6	69.8

FIGURE 4.2.1. Unirradiated Reference Fuel Assemblies

BWR fuel models that account for all radionuclides in the fuel and take into account differences in fuel exposures for PWR and BWR fuel assemblies and the effects of reduced exposure for startup and shutdown cores.

The amounts of some selected radionuclides present in the example fuel composition are listed in Table 4.2.2. These radionuclides were selected based on several factors, among which are 1) potential for release, 2) potential effect of release, 3) quantity present, and 4) public interest. These nuclides and their radioactive daughter nuclides provide most of

TABLE 4.2.2. Selected Radionuclide Content in Example Once-Through Cycle Spent Fuel

		Ci/GWe-yr for Various Decay Periods				
Fission Products[a]		1.5 yr	5 yr[b]	10 yr	50 yr	100 yr
^{3}H	(1.2×10^{1})	1.6×10^{4}	1.3×10^{4}	9.5×10^{3}	1.0×10^{3}	6.1×10^{1}
^{85}Kr	(1.1×10^{1})	3.4×10^{5}	2.7×10^{5}	1.9×10^{5}	1.5×10^{4}	6.1×10^{2}
^{90}Sr	(2.9×10^{1})	2.5×10^{6}	2.2×10^{6}	2.0×10^{6}	7.4×10^{5}	2.2×10^{5}
^{106}Ru	(1.0)	6.5×10^{6}	3.8×10^{5}	1.3×10^{4}		
^{129}I	(1.6×10^{7})	1.3	1.3	1.3	1.3	1.3
^{134}Cs	(2.1)	4.6×10^{6}	1.2×10^{6}	2.2×10^{5}	2.9×10^{-1}	
^{137}Cs	(3.0×10^{1})	3.5×10^{6}	3.2×10^{6}	2.9×10^{6}	1.1×10^{6}	3.6×10^{5}
^{144}Ce	(7.8×10^{-1})	9.5×10^{6}	2.7×10^{5}	3.1×10^{3}		
Total all Fission Products		5.3×10^{7}	1.6×10^{7}	1.0×10^{7}	3.7×10^{6}	1.1×10^{6}
Actinides[a]						
^{238}U	(4.5×10^{9})	1.2×10^{1}	1.2×10^{1}	1.2×10^{1}	1.2×10^{1}	1.2×10^{1}
^{238}Pu	(8.9×10^{1})	8.0×10^{4}	7.9×10^{4}	7.6×10^{4}	5.6×10^{4}	3.8×10^{4}
^{239}Pu	(2.4×10^{4})	1.1×10^{4}	1.1×10^{4}	1.1×10^{4}	1.1×10^{4}	1.1×10^{4}
^{240}Pu	(6.8×10^{3})	1.7×10^{4}	1.7×10^{4}	1.7×10^{4}	1.7×10^{4}	1.7×10^{4}
^{241}Pu	(1.3×10^{1})	4.2×10^{6}	3.4×10^{6}	2.6×10^{6}	4.1×10^{5}	3.8×10^{3}
^{241}Am	(4.6×10^{2})	1.4×10^{4}	3.5×10^{4}	6.1×10^{4}	1.3×10^{5}	1.3×10^{5}
^{242}Cm	(4.5×10^{-1})	1.4×10^{5}	6.0×10^{2}	3.2×10^{2}	2.7×10^{2}	2.1×10^{2}
^{244}Cm	(1.8×10^{1})	4.9×10^{4}	4.2×10^{4}	3.4×10^{4}	7.4×10^{3}	1.1×10^{3}
Total All Actinides		4.5×10^{6}	3.6×10^{6}	2.9×10^{6}	6.3×10^{5}	2.4×10^{5}
Activation Products[a]						
^{14}C	(5.7×10^{3})	2.8×10^{1}	2.8×10^{1}	2.8×10^{1}	2.8×10^{1}	2.8×10^{1}
^{55}Fe	(2.4)	1.6×10^{5}	3.8×10^{4}	1.3×10^{4}	2.7×10^{-1}	4.5×10^{-7}
^{60}Co	(5.3)	1.6×10^{5}	1.1×10^{5}	4.0×10^{4}	2.6×10^{3}	2.8×10^{-1}
^{63}Ni	(9.2×10^{1})	1.5×10^{4}	1.5×10^{4}	1.5×10^{4}	1.3×10^{4}	7.7×10^{3}
Total All Activation Products		3.5×10^{5}	2.1×10^{5}	7.2×10^{4}	1.6×10^{4}	7.7×10^{3}

(a) Numbers in parentheses are the half-lives (in years).
(b) A minimum age of 5 yr is assumed here for shipment of spent fuel from the reactors in the once-through cycle.

the radioactivity in spent fuel while predisposal operations take place. Tables in Appendix A of Volume 2 provide data for these and other radionuclides for longer time periods.

Substantial quantities of non-TRU wastes are generated in the once-through fuel cycle during operation of nuclear power plants and spent fuel storage facilities. Depending on the treatment in the once-through fuel cycle, substantial amounts of TRU secondary wastes may or may not be produced. If the treatment mode involves simply the packaging of intact spent fuel, the secondary waste produced in the packaging operation should contain very little TRU radioactivity and is considered here to be all non-TRU waste. However, if the spent fuel cladding is breached in the treatment process, then secondary TRU wastes would be produced. Depending on the complexity of such a process, substantial amounts of TRU secondary waste could be produced. The secondary TRU wastes from the once-through fuel cycle would be similar to some of the primary wastes in the reprocessing case.

4.2.2 Reprocessing Cycle

When spent fuel is processed to recover (for recycle) the uranium and plutonium it contains, primary TRU wastes of two types are generated in recycle facilities: 1) fuel reprocessing plant (FRP) wastes and 2) mixed oxide fuel fabrication plant (MOX-FFP) wastes. In fuel reprocessing plants the spent fuel is dissolved out of the cladding, the uranium and plutonium are recovered and purified by a series of solvent extraction operations, and the uranium and plutonium products are converted to UF_6 and PuO_2 (or mixed UO_2-PuO_2) for further use. In mixed oxide fuel fabrication plants the PuO_2 (or mixed UO_2-PuO_2) is blended with UO_2, processed to a suitable form, and incorporated into mixed oxide (MOX) fuel elements to be recycled to a nuclear power plant. More extensive descriptions of such facilities are presented in DOE/ET-0028 (Section 3.2).

Table 4.2.3 contains the estimated quantities and selected radionuclide contents of the primary high-level, TRU, and gaseous wastes generated in the reprocessing cycle. The radionuclide contents are given as fractions of the amounts present in the recycle spent fuel for the FRP wastes and as fractions of the amounts present in the fabricated MOX fuel for the MOX FFP wastes. These amounts are presented in Table 4.2.4, for an example recycle spent fuel, and in Table 4.2.5, for an example MOX fuel. Except for the isotopes of uranium and plutonium, the total amounts of radionuclides present in the untreated wastes of the two fuel cycles may be directly compared using the data of Tables 4.2.2 and 4.2.4. The quantities of uranium and plutonium in the reprocessing cycle wastes amount to about 1% of that present in the spent fuel.

Wastes from two areas of the fuel reprocessing plant (the fuel storage basin and the uranium conversion facility) are classified as non-TRU wastes. As in the once-through cycle, non-TRU wastes also result from operation of nuclear power plants and spent fuel storage facilities.

TABLE 4.2.3. Selected Radionuclide Content in Primary High-Level, TRU, and Gaseous Wastes from Fuel Reprocessing Plant and MOX Fuel Fabrication Plant

			Nuclide Content in Waste Category(b) as a Fraction of that Present in Spent Fuel (c) or in MOX Fuel(d)												
			Fission Products						Actinides			Activation Products			
Waste Category	Facility	Volume,(a) m^3/GWe-yr	H	Kr	Sr, Cs	Ru	I	Ce	Pu	Am	Cm	C	Fe	Co	Ni
High-Level Liquid Waste	FRP	22	0.08	0	1	1	5×10^{-3}	1	5×10^{-3}	1	1	0	0	0	0
Fuel Residue	FRP														
Hulls		10	0.15	0	5×10^{-4}	5×10^{-4}	0	5×10^{-4}	5×10^{-4}	5×10^{-4}	5×10^{-4}	0.09	0.02	0.04	0.01
Hardware		2.1	0	0	0	0	0	0	0	0	0	0.07	0.98	0.96	0.99
Failed Equipment	FRP	8.4	0	0	1×10^{-6}	1×10^{-6}	0	1×10^{-6}	1×10^{-4}	1×10^{-6}	1×10^{-6}	0	0	0	0
	MOX FFP	1.5	0	0	0	0	0	0	1×10^{-5}(d)	1×10^{-6}(d)	0	0	0	0	0
Noncombustible Waste	FRP	15	0	0	1×10^{-6}	1×10^{-6}	0	1×10^{-6}	1×10^{-4}	1×10^{-6}	1×10^{-6}	0	0	0	0
	MOX FFP	1.5	0	0	0	0	0	0	1×10^{-4}(d)	1×10^{-4}(d)	0	0	0	0	0
Compactable and Combustible Waste															
Trash • Process Matl's	FRP	62	0	0	1.1×10^{-6}	1.1×10^{-5}	2.1×10^{-3}	1.1×10^{-6}	6.1×10^{-4}	1.1×10^{-6}	1.1×10^{-6}	0	0	0	0
	MOX FFP	3.8	0	0	0	0	0	0	3×10^{-4}(d)	3×10^{-4}(d)	0	0	0	0	0
Filters	FRP	6.1	0	0	1×10^{-5}	1×10^{-5}	0	1×10^{-5}	2×10^{-3}	1×10^{-5}	1×10^{-5}	0	0	0	0
	MOX FFP	0.76	0	0	0	0	0	0	7×10^{-4}(d)	7×10^{-4}(d)	0	0	0	0	0
Concentrated Liquids, Wet Wastes, and Particulate Solids	FRP	5.7	1×10^{-3}	0	1×10^{-5}	1×10^{-3}	3×10^{-3}	1×10^{-5}	1×10^{-3}	1×10^{-5}	1×10^{-5}	0	0	0	0
	MOX FFP	2.8	0	0	0	0	0	0	1.1×10^{-4}(d)	2×10^{-2}(d)	0	0	0	0	0
Gaseous Wastes															
Dissolver Off-Gas	FRP	2.6×10^{4}(b)	0.05	1	1×10^{-7}	2×10^{-4}	1	1×10^{-7}	1×10^{-7}	1×10^{-7}	1×10^{-7}	0.84	0	0	0
Vessel Off-Gas	FRP	1.1×10^{6}(b)	1×10^{-3}	1×10^{-6}	1×10^{-7}	1×10^{-7}	5×10^{-3}	1×10^{-7}	1×10^{-7}	1×10^{-7}	1×10^{-7}	0	0	0	0
Vaporized Excess Water	FRP	7.6×10^{5}(b)	0.72	1×10^{-10}	1×10^{-16}	1×10^{-10}	1×10^{-5}	1×10^{-16}	1×10^{-11}	1×10^{-16}	1×10^{-16}	0	0	0	0
	MOX FFP	1.5×10^{4}(b)	0	0	0	0	0	0	1×10^{-12}(d)	1×10^{-10}(d)	0	0	0	0	0
Ventilation Air	FRP	1.2×10^{8}(b)	1×10^{-11}	1×10^{-11}	1×10^{-11}	1×10^{-11}	1×10^{-11}	1×10^{-11}	1×10^{-11}	1×10^{-11}	1×10^{-11}	0	0	0	0
	MOX FFP	8.4×10^{6}(b)	0	0	0	0	0	0	3×10^{-7}(d)	3×10^{-7}(d)	0	0	0	0	0

(a) Data obtained from Section 4 of DOE/ET-0028.
(b) Data obtained form Section 3.3 of DOE-ET-0028.
(c) Quantities present in spent fuel are listed in Table 4.2.4.
(d) Quantities present in MOX fuel are listed in Table 4.2.5.

TABLE 4.2.4. Selected Radionuclide Content in Example Recycle Spent Fuel

Fission Products	Ci/GWe-yr for Various Decay Periods 1.5 yr[a]	6.5 yr[b]	10 yr	50 yr	100 yr
^{3}H	1.6×10^{4}	1.2×10^{4}	9.9×10^{3}	1.0×10^{3}	6.1×10^{1}
^{85}Kr	3.2×10^{5}	2.3×10^{5}	1.8×10^{5}	1.4×10^{4}	5.7×10^{2}
^{90}Sr	2.3×10^{6}	2.2×10^{6}	1.9×10^{6}	6.9×10^{5}	2.0×10^{5}
^{106}Ru	7.2×10^{6}	2.2×10^{5}	1.4×10^{4}		
^{129}I	1.3	1.3	1.3	1.3	1.3
^{134}Cs	4.6×10^{6}	8.4×10^{5}	2.2×10^{5}	2.9×10^{-1}	
^{137}Cs	3.5×10^{6}	3.2×10^{6}	2.9×10^{6}	1.1×10^{6}	1.1×10^{6}
^{144}Ce	9.1×10^{6}	1.1×10^{5}	3.0×10^{3}		
Total All Fission Products	5.3×10^{7}	1.3×10^{7}	1.0×10^{7}	3.6×10^{6}	1.1×10^{6}
Actinides					
^{238}U	1.2×10^{1}	1.2×10^{1}	1.2×10^{1}	1.2×10^{1}	1.2×10^{1}
^{238}Pu	2.1×10^{5}	2.1×10^{5}	2.0×10^{5}	1.5×10^{5}	9.9×10^{4}
^{239}Pu	1.4×10^{4}	1.4×10^{4}	1.4×10^{4}	1.4×10^{4}	1.4×10^{4}
^{240}Pu	2.8×10^{4}	2.8×10^{4}	2.8×10^{4}	2.8×10^{4}	2.8×10^{4}
^{241}Pu	6.8×10^{6}	5.3×10^{6}	4.6×10^{6}	6.7×10^{5}	6.5×10^{4}
^{241}Am	2.7×10^{4}	7.6×10^{4}	1.0×10^{5}	2.2×10^{5}	2.2×10^{5}
^{242}Cm	3.8×10^{5}	1.6×10^{3}	1.4×10^{3}	1.2×10^{3}	9.5×10^{2}
^{244}Cm	2.7×10^{5}	2.2×10^{5}	1.9×10^{5}	4.2×10^{4}	6.1×10^{3}
Total All Actinides	7.6×10^{6}	5.7×10^{6}	5.1×10^{6}	1.1×10^{6}	4.6×10^{5}
Activation Products					
^{14}C	2.1×10^{1}	2.1×10^{1}	2.1×10^{1}	2.1×10^{1}	2.1×10^{1}
^{55}Fe	1.6×10^{5}	3.8×10^{4}	1.3×10^{4}	2.7×10^{-1}	4.5×10^{-7}
^{60}Co	1.6×10^{5}	8.0×10^{4}	4.0×10^{4}	2.6×10^{3}	2.8×10^{-1}
^{63}Ni	1.5×10^{4}	1.5×10^{4}	1.5×10^{4}	1.3×10^{4}	7.7×10^{3}
Total All Activation Products	3.5×10^{5}	1.3×10^{5}	7.2×10^{4}	1.3×10^{4}	7.7×10^{3}

(a) A minimum age of 1.5 yr is assumed here for reprocessing.
(b) A minimum age of 6.5 yr is assumed here for shipment of solidified high-level waste.

TABLE 4.2.5. Selected Radionuclide Content in Example MOX fuel

Actinides	Ci/GWe-yr[a] for Different Times Since Reprocessing: 1 yr[b]	10 yr
^{238}Pu	1.4×10^5	1.4×10^5
^{239}Pu	9.9×10^3	9.9×10^3
^{240}Pu	2.0×10^4	2.0×10^4
^{241}Pu	4.4×10^6	2.9×10^6
^{241}Am	7.3×10^3	5.9×10^4
Total	4.6×10^6	3.1×10^6

(a) Assuming 20% of fuel to reactors is recycle MOX fuel.
(b) A period of 1 yr is assumed here between reprocessing and MOX fuel fabrication.

4.3 WASTE TREATMENT AND PACKAGING

This section addresses the treatment and packaging of high-level (including spent fuel), TRU, and gaseous wastes resulting from the once-through and the reprocessing cycles. The principal source of the information contained herein is DOE/ET-0028, Technology for Commercial Radioactive Waste Management (DOE 1979), which was prepared in support of this Statement.[a] The processes described here are not necessarily optimized but are representative of currently available technology.

The treated waste form and container each provide a barrier to release of radionuclides after disposal. The functions of the treated waste forms and containers are discussed in more detail in Section 5.1.2.

4.3.1 Spent Fuel Treatment and Packaging in Once-Through Cycle

In the once-through fuel cycle, the spent fuel is considered to be waste and is treated to prepare it for disposal. Treatment processes that have been examined range from simply 1) packaging the intact spent fuel assemblies to 2) chopping the fuel assemblies to expose the fuel, utilizing a process called voloxidation to remove a portion of the volatile radionuclides, dissolving the fuel in nitric acid and finally converting the solution to a solid by calcination and vitrification.

Encapsulation of intact spent fuel assemblies for geologic disposal is the example process assumed in this Statement for the once-through fuel cycle. Three other treatment methods are also described to illustrate the range of treatment alternatives available.

4.3.1.1 Encapsulate Intact Assembly (Example Method)

A detailed description of the example encapsulation process is contained in DOE/ET-0028 (Section 5.7.3). A similar process is described in ONWI-39 (Appendix C). In both of these process concepts the intact fuel assemblies are placed in steel canisters that are then backfilled with helium and welded closed. A flow diagram for the process is shown in Figure 4.3.1.

The canister and filler materials included in the studies discussed here are only a few of the potentially applicable materials. Canister materials being considered by DOE include a variety of metal alloys, ceramics, carbides, forms of carbon, glasses, and cements; potential filler (stabilizer) materials include a variety of gases, castable solids, and granular

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(a) Additional once-through cycle concepts were discussed later in "An Assessment of LWR Spent Fuel Disposal Options," ONWI-39 (ONWI 1979); this report also contains information on a reprocessing case which is somewhat different in waste treatment philosophy than that presented in DOE/ET-0028. Other recent descriptions of reprocessing waste treatment operations are contained in "Design Integration Study, Spent LWR Fuel Recycle Complex, Conceptual Design, Case A-1, Separated Streams," DP-CFP-78-121 (SRL 1978) and "Design Integration Study, Spent LWR Fuel Recycle Complex, Conceptual Design, Coprocessing Case A-2," DP-CFP-121-79 (Harries et al. 1979). Various methods of waste treatment and packaging for both fuel cycles are also addressed in "Technical Support of Standards for High-Level Radioactive Waste Management, Volume B, Engineering Controls," EPA 520/4-79-007B (EPA 1977).

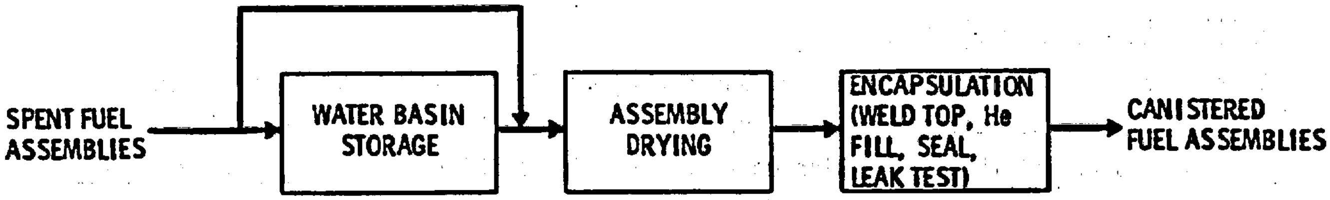

FIGURE 4.3.1. Flow Diagram for Encapsulation of Intact Spent Fuel Assemblies

solids (DOE/NE-0007, Section II.E.1). The waste package finally chosen will be tailored to the geologic environment in which the package is to be disposed.

In the DOE/ET-0028 study, the cleaned and dried fuel assemblies are individually packaged in square canisters[a] that are only slightly larger than the assemblies themselves. A canister for a PWR assembly has dimensions of 0.24 x 0.24 x 4.88 m (9.5 x 9.5 x 192 in.) and a canister for a BWR assembly has dimensions of 0.165 x 0.165 x 4.88 m (6.5 x 6.5 x 192 in.). For the mixture of fuel used in this generic study (40% of the assemblies are from PWRs and 60% are from BWRs), 127 canisters are filled per GWe-yr.

The process concept described in ONWI-39 (Appendix C) is very similar except that cylindrical canisters are used, and the BWR assemblies are packaged three to a canister. A canister for a PWR assembly has dimensions of 0.36 x 4.72 m (14 x 186 in.) and a canister for three BWR assemblies has dimensions of 0.41 x 4.72 m (16 x 186 in.). Seventy-eight canisters per GWe-yr are required in this instance for the mixture of fuel used in this generic study.

The DOE/ET-0028 and the ONWI-39 studies present different estimates of TRU waste produced during the treatment operations. DOE/ET-0028 concluded that waste produced during the treatment of the intact fuel assemblies could be considered to be non-TRU (as is waste produced during the irradiation and the subsequent storage of the assemblies). ONWI-39, however, lists appreciable quantities of TRU wastes resulting from packaging of the intact assemblies (but does not say in which operations they arise). The actual amount remains to be determined from operating experience; if a significant amount of TRU waste is indeed generated during the packaging of intact spent fuel, then the spent fuel capacity of the repositories described in Chapter 5 may be somewhat overstated.

Consideration is also given in ONWI-39 (Section 10.3) to other canister design variations. Alternative canister void filler materials considered include gases other than helium (e.g., air, nitrogen, or argon), monolithic solid fillers formed by pouring molten materials (e.g., lead, aluminum, or glass) into the canister and then cooling, and granular solid fillers (e.g., lead shot, sand, or glass frit). The use of thicker walls in the primary canisters, overpacks, and increasing the number of spent fuel assemblies per canister were also considered.

(a) Square canisters allow a more close-packed array during interim storage but are not as strong as cylindrical canisters with the same wall thickness.

Another variation considered in ONWI-39 (Section 10.6) involves disassembly prior to packaging so that the canisters contain spent fuel rods only, instead of complete assemblies. In this option the end fittings are removed from the fuel elements, the elements are disassembled, and the fuel rods are bundled together and sealed into canisters.

4.3.1.2 Chop Fuel Assembly, Voloxidize Fuel, and Encapsulate

A process for chopping the fuel assemblies, removing volatile components through voloxidation, and encapsulating the spent fuel is described in ONWI-39 (Appendix C). The end fittings of the spent fuel are first cut off and encapsulated. The remaining portions of the fuel assemblies are then chopped and voloxidized, and encapsulated in canisters. A flow diagram for the process is shown in Figure 4.3.2.

The voloxidation process, which is in the development stage (Groenier 1977), promotes the release of gaseous fission products from the fuel by oxidizing UO_2 to U_3O_8 at 400°to 500°C in air. This oxidation results in disintegration of the fuel, which provides an easier escape path for the gaseous fission products. Removal of the gaseous fission products from the off-gas stream is addressed in Section 4.3.4.

The processed spent fuel is encapsulated in cylindrical steel canisters that are helium-filled, sealed by welding, and leak tested. Any leaking canisters are overpacked in a second larger canister. The primary canister size is 0.30 x 3.0 m (12 x 120 in.). Sixty-one canisters per GWe-yr are estimated to be required to contain the chopped and voloxidized fuel.

The end fittings sheared from the fuel-bearing portions of spent fuel are packaged without further processing in 0.5 x 3.0 m cylindrical canisters. One canister holds the ends of either three PWR or six BWR assemblies; for the mixture of fuel used in this generic study, 11.6 canisters are filled per GWe-yr.

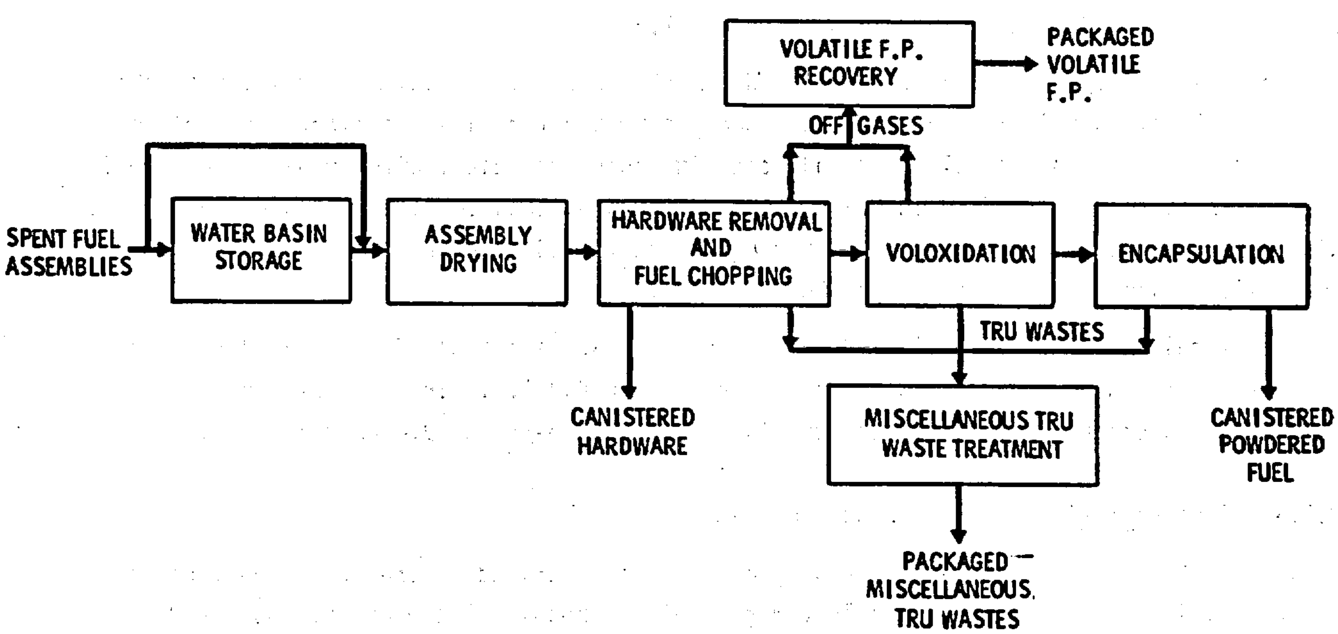

FIGURE 4.3.2. Flow Diagram for Encapsulation of Chopped and Voloxidized Spent Fuel

Combustible wastes produced during the processing (secondary wastes) are converted to ashes in an incinerator, and the ashes are blended with fixation materials and placed into waste containers. Incineration is accomplished in a molten salt combustion unit followed by fixation of TRU ashes in aluminum silicate mineral (clay). Noncombustible secondary wastes are also blended with fixation materials and placed into waste containers. Large pieces of failed equipment are disassembled or cut into smaller pieces suitable for packaging. The wastes requiring remote handling are packaged in 0.5 x 3.0-m cylindrical canisters, and the wastes suitable for contact handling are packaged in 55-gallon drums. The estimated numbers of these secondary waste packages considered to be TRU wastes are 30 canisters/GWe-yr and 6.5 drums/GWe-yr.

4.3.1.3 Dissolve Fuel and Convert to Glass

A process for dissolution of fuel and conversion to glass is described in ONWI-39 (Appendix C). This process incorporates fuel chopping and dissolution followed by concentration and calcination of the resultant solution followed by vitrification (conversion to glass) of the calcine. Voloxidation of the chopped fuel is also included in the process, as described in Section 4.3.1.2. A flow diagram for this process is shown in Figure 4.3.3. Although glass is the waste form described in ONWI-39, other waste forms such as those discussed in Section 4.3.2 could also be employed.

The voloxidized fuel is dissolved in nitric acid. During this operation the portions of the iodine and krypton that were not released to the off-gas system during voloxidation are evolved. The off-gas treatment process is described in Section 4.3.4.

The dissolution process also allows separation of the fuel cladding hulls from the fuel itself. The hulls are compacted in small containers with a hydraulic press and several of these containers are banded together and placed in a 0.5 x 3.0-m cylindrical canister. The required number of such canisters is estimated to be 17.5 per GWe-yr. The fuel assembly end fittings are packaged as described in Section 4.3.1.2.

The dissolved spent fuel is concentrated and then spray-calcined. The calcine is then fed along with glass frit into a continuous ceramic melter for vitrification. The molten glass that emerges from the melter is collected in canisters which, after cooling, are seal-welded. The referenced study uses 0.5 x 3.0-m cylindrical canisters; the number required is estimated to be 141 per GWe-yr. The number of canisters will vary however, depending on the thermal limitations of the final repository.

The miscellaneous combustible and noncombustible wastes and the failed equipment are treated the same as in the process described in Section 4.3.1.2. The estimated numbers of the TRU waste packages in this process are 43 canisters/GWe-yr and 9.4 drums/GWe-yr.

4.3.1.4 Dissolve Fuel for Disposal as a Liquid

The spent fuel treatment and packaging operations described in the preceding three sections result in waste packages suitable for geologic disposal. These operations could doubtless be adapted to provide different packages (if required) for disposal by some of the

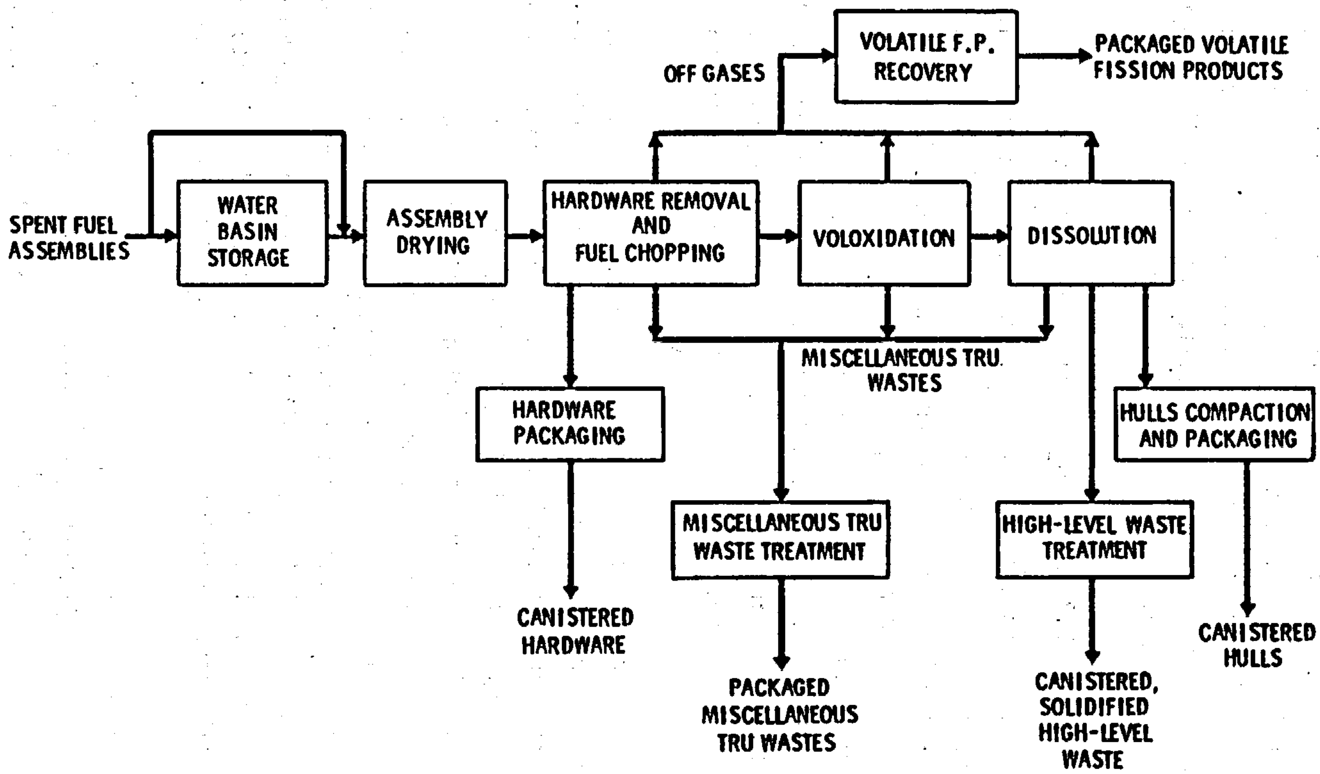

FIGURE 4.3.3. Flow Diagram for Encapsulation of Dissolved and Vitrified Spent Fuel

methods described in Chapter 6 as alternatives to geologic disposal. However, two of these alternative disposal methods (rock melting and well injection) involve disposal of the high-level waste in liquid form; thus, a modified spent fuel treatment process is required. Application of these methods to disposal of dissolved spent fuel presents added nuclear criticality safety problems and feasability uncertainties resulting from the presence in the solution of all of the plutonium and the uranium.

By eliminating the calcination and vitrification operations, the spent fuel treatment process described in Section 4.3.1.3 could provide a liquid waste stream for disposal. Additional storage would probably have to be provided for the dissolved spent fuel solution to allow proper operation of the disposal process, however. A flow diagram for such a process is shown in Figure 4.3.4.

4.3.2 High-Level Liquid Waste Treatment

High-level liquid wastes are defined as "those aqueous wastes resulting from the operation of the first cycle solvent extraction system, or equivalent, and the concentrated wastes from subsequent extraction cycles, or equivalent, in a facility for reprocessing irradiated reactor fuels" (10 CFR 50). These wastes contain over 99% of the nonvolatile fission products and actinides, except U and Pu. If spent fuel is reprocessed, the U and Pu will normally be recycled. Only a small amount of U and Pu, perhaps 0.5%, resulting from waste losses during reprocessing will be in the HLW. Liquid high-level waste can be stored in tanks as an interim measure, but it must be solidified before transportation and disposal.

Many HLW treatment processes are under development and DOE is committed to examining the relative merits of many of these processes. For this discussion the candidate processes have been divided into three categories: those that convert the HLW into glass (Section 4.3.2.2), into a crystalline solid (Section 4.3.2.3), or into a composite or multiphase solid form (Section 4.3.2.4). A further important distinction concerning the candidate HLW waste treatment processes should also be made. The processes fall into two broad classes: those that have been developed to the stage of practical engineering-scale implementation, and those for which there has been some characterization of waste form properties but little or no process development. Calcine, low-melting glass and cement can be placed in the first category. All of the rest of the waste forms to be described fall into the latter, relatively undeveloped category. Additional data on many of these processes may be found in ERDA-76-43.

The processing descriptions given here assume that the HLW is not partitioned before treatment; however, because chemical partitioning has potential as a pretreatment for high-level liquid waste, partitioning techniques are also discussed in this section.

Before proceeding with the more general discussion, brief descriptions will be given of the two well developed high-level liquid waste treatment processes used in this Statement for evaluation of environmental impacts and costs. These processes are:

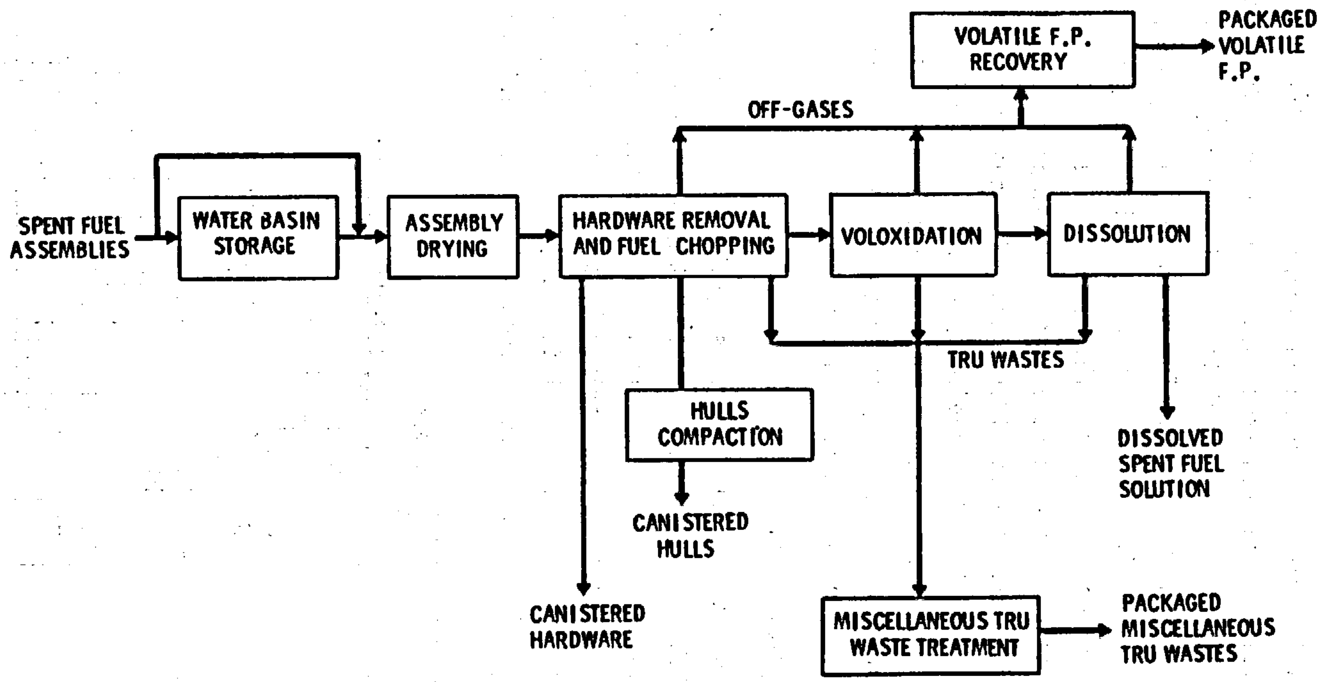

FIGURE 4.3.4. Flow Diagram for Dissolution of Spent Fuel for Disposal as a Liquid

1) vitrification by in-can melting following spray calcination and 2) fluidized bed calcination. These processes are described in detail in DOE/ET-0028. They produce a borosilicate glass product and a granular powder product, respectively.

Spray Calciner/In-Can Melting (Example Method)

A flow diagram for the in-can melting process, the example high-level waste solidification process of this Statement, is shown in Figure 4.3.5. The liquid HLW is dried and calcined in a spray-calciner, the resultant calcine is mixed with about twice its weight of glass-forming materials, and the mixture is melted within a steel canister. The filled canister is cooled and sealed by welding. The output of the example process amounts to about 2.2 m^3 of waste glass per GWe-yr; higher volumes would result from lower waste loadings. The number of canisters used to contain this volume of glass depends on a number of factors, among which are the heat generation rate of the contained waste and the heat generation rate per canister allowed by disposal considerations. For canister heat loadings of 1.2 to 3.2 kW (typical of those allowed in geologic repositories) and 6.5-year aged (out-of-reactor) waste, the number of canisters would amount to 44 and 17, respectively, per GWe-yr. A large variety of other glass-making processes have been developed; the output of these processes would be similar to that described here.

Fluidized Bed Calcination

In the fluidized bed calcination process (other calcination processes are also feasible), the liquid HLW is atomized as it enters the calciner vessel, which is heated by an in-bed combustion system. When the atomized HLW is injected into the hot bed, the waste constituents are converted to solids (primarily oxides) that adhere to the surface of particles already in the bed. The bed is fluidized by heated air entering through perforations in the bottom support plate. Calcined product is removed continuously so that the bed

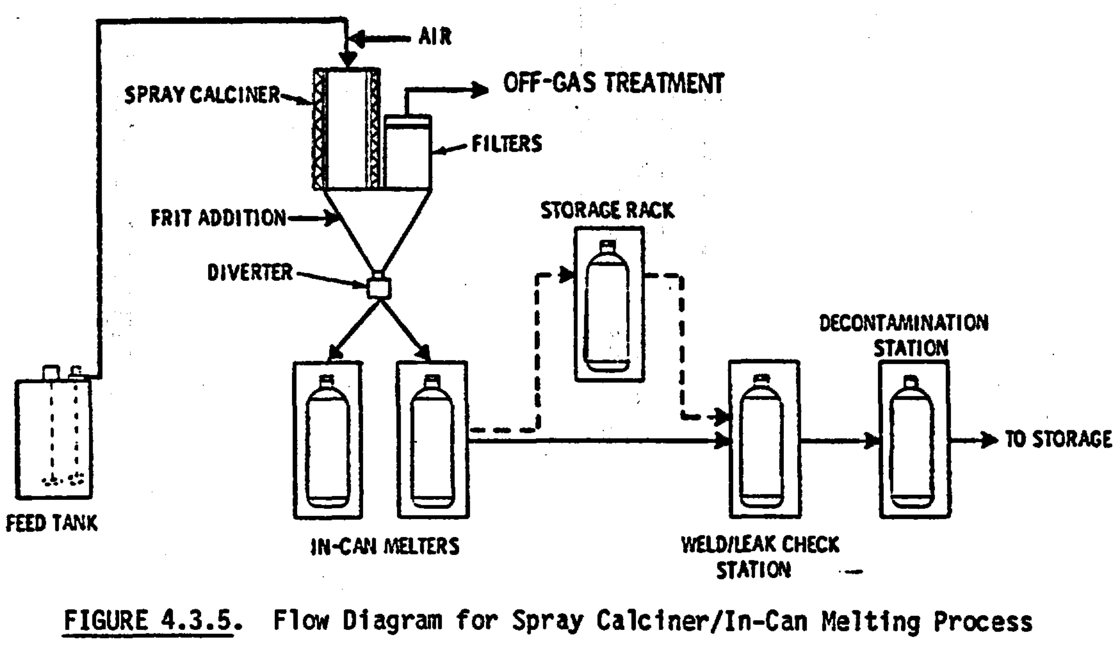

FIGURE 4.3.5. Flow Diagram for Spray Calciner/In-Can Melting Process

inventory remains essentially constant. The calcine is collected in canisters and residual water and nitrate are removed by heating to 700°C before the canisters are sealed shut. The output of the example process amounts to about 0.9 m^3 of calcine per GWe-yr. A smaller diameter canister may be required for waste calcine than for waste glass to prevent overheating at the centerline of the canister, because of the lower thermal conductivity of calcine.

4.3.2.1 Chemical Partitioning

The partitioning or separation of certain elements from nuclear fuel cycle wastes has been viewed as a potential means for improving waste management (ERDA 1976, Campbell 1976, Schneider and Platt 1974, Cooperstein 1976). The perceived benefits result from removal of certain radionuclides and, hence, improvements in the management of the resulting partitioned radionuclide fraction compared to the management options for the unpartitioned wastes. Three subsequent options for disposal of partitioned radionuclides are discussed in this document: 1) transmutation as discussed in Section 6.1.7, 2) chemical resynthesis as discussed in Section 4.3.2.3, and 3) space disposal as discussed in Section 6.1.8.

In general, to partition simply means to separate elements, or groups of elements, from some mixture of chemical species. In a nuclear fuel cycle, partitioning would occur mainly during the reprocessing of spent fuel (ERDA 1976). There are many chemical elements in spent nuclear fuels (see Section 2.1), and many combinations in which these elements may be chemically separated. Consequently, there are also numerous partitioning alternatives that may facilitate useful waste treatment alternatives or disposal options. For all the specific partitioning candidates described here, one must realize that: 1) no partitioning processes have been demonstrated for waste disposal on a commercial scale; 2) historically most recovery processes leave several percent, or more, of the desired elements in the waste streams; and 3) partitioning for waste management purposes requires substantially higher recoveries than have been achieved to date. Partitioning itself is not an option for final disposal of radioactive wastes, although some waste partitioning may be required as a pretreatment to permit the final disposal of the resulting waste fraction (e.g., the partitioning of fission product iodine for space disposal).

With respect to waste management, partitioning may lead to improved waste characteristics for either the short term (less than 1000 years) or the long term (greater than 1000 years). The partitioning of strontium and cesium, for example, may be a useful option to reduce the self-heating (Buckingham 1967) characteristics of high-level wastes over the short term and thereby permit the storage of salt cakes that are not overly self-heating. In addition, the partitioning of actinides as well as some fission products may be useful to reduce the long-term radiotoxicity of wastes (Bond and Leuze 1975, Croff et al. 1977) and, therefore, reduce the exposure of future populations to radioactivity should the wastes ever be reintroduced into man's environment in the distant future (say after 100,000 years of storage).

Some partitioning options may be useful for maximizing energy conservation in the fuel cycle, facilitating the beneficial use (Rohrmann 1968) of selected fission products, and improving nuclear safeguards (Campbell and Gift 1978; Pobereskin, Kok and Madia 1977). The recovery of cesium, for example, has been examined for use in sterilizing sewage sludges (Sivinski 1975; Reynolds, Hagengruber and Zuppero 1974); strontium might also be used as a heat source (Dix 1975) in remote and inacccessible areas. Partitioned palladium, rhodium, ruthenium and technetium could become mineral resources.

On the other hand, partitioning will invariably complicate waste management during the operation of the fuel cycle, as compared with other existing methods of dealing with the unpartitioned wastes (ERDA-76-43, Section 16.2). Several reasons for this are:

- Increased production of secondary wastes. Althought the chemistry associated with the partitioning of radionuclides is quite diverse, all known options generate significant quantities of secondary wastes that must be managed. These secondary wastes may be treated by incineration, by compaction, by immobilization, or by other methods, but invariably the waste volumes will be increased by the partitioning, and waste management costs will also increase. Many partitioning options will significantly increase the high-level waste volume because of the addition of salting agents or other nonvolative species. Also, many chemical additives may adversely affect high-level waste solidification and the long-term stability of the waste form (e.g., glass devitrification).
- Increased transportation costs and requirements. Most partitioned waste fractions can be transported safely only with extensive shielding. For many of the transmutation cycles, the transmutable elements are recycled many times before a significant reduction in quantity is achieved. In the case of actinides some of the transmuted products are strong neutron emitters and will constitute a handling problem.
- Increased costs due to partitioning and secondary waste treatment. All known partitioning options involve sophisticated chemical separation processes that must be remotely maintained and operated. Significant capital investment and operating costs will result if these chemical processes are implemented. The recovery of selected waste constituents, like cesium and strontium, does not significantly reduce the cost of managing the residual high-level waste.
- Increased potential for radiation exposure. Since partitioning will require increased chemical operations, handling, transportation, and storage, the potential for increased occupational radiation exposure also exists. The potential for accidental release of radioactive material (and general population exposure) will also be increased. These factors must be quantified if partitioning is adopted.
- Increased thermal loading. Partitioned waste fractions with high heat generation densities impose a higher thermal load on containment materials than does unpartitioned waste. A recent study (NAS 1978) has suggested that the permanent containment of cesium and strontium partitioned from wastes at Hanford will be difficult because of the high heat densities involved.

4.3.2.2 Glass Waste Forms

Vitrification (conversion to glass) of high-level liquid wastes is being developed in Germany, France, India, Russia, Great Britain, Belgium, Japan, Canada and the United States. A facility for vitrification of the HLW from the Marcoule reprocessing plant has been operating in France since the summer of 1978 (Bonniand et al. 1978). The various HLW vitrification processes and properties of the glasses made by them have been well described in recent reports and symposia proceedings (McCarthy 1979, Chikalla and Mendel 1979).

Low-Melting Glasses

Low-melting glasses are glasses that can be processed at temperatures below about 1200°C. The most well developed vitrification processes throughout the world all produce low-melting glasses of a borosilicate formulaion, although a small amount of development continues on phosphate glass formulations (Kelley 1975, Wiley and LeRoy 1979, Gombert et al. 1979, Kupfer 1979 and Mendel 1978). The product of these borosilicate glass processes is a glass casting in a metal canister. The castings vary in size depending on the process and the amount of radioactivity, but are generally cylinders from 0.3 to 0.6 m in diameter and 1 to 3 m long.

Borosilicate waste glasses can contain one-third or more (by weight) HLW oxides; the remainder is inert glass-forming material added during vitrification processing. The glasses can tolerate wide variations in HLW composition without sacrificing their properties. The glass castings contain some fractures caused by thermal stresses induced as the large monoliths cool. Waste glasses are metastable materials and they must be cooled fairly rapidly (a cooling rate of at least 10°C/hr between 900°C and 600°C is satisfactory for most formulations) to prevent excessive devitrification from occurring. At lower temperatures, e.g., those encountered in geologic disposal, the rates of thermal devitrification are too slow to be a factor. Extensive studies have shown that the only significant effect of devitrification, if it does occur, is a small increase in leach rate. The increase is usually less than a factor of three even in fully devitrified glasses but in some formulations may be as high as 10. The glass phase exhibits excellent stability in radiation fields as shown by tests simulating over 500,000 years of alpha radiation.

Borosilicate waste glasses also exhibit good chemical durability; however, there is a finite reaction rate with water. The reaction rate is dependent on many factors but for typical waste glasses is usually in the range 10^{-7} to 10^{-5} g glass/cm^2-day after a few weeks of leaching at 25°C. The rate increases with temperature, rising a factor of 10 to 100 for a 100°C increase in temperature.

High-Temperature Glasses

In the context of this discussion, these are glasses that melt above 1200°C. They contain more silica or alumina than the low-temperature glasses. An early example of a high-temperature waste glass is the nepheline syenite waste glass made in Canada from 1958 to 1960. Blocks of this glass, without canisters, were buried below the water table at Chalk River in 1960. The leaching behavior of these glass blocks has been monitored by means of wells. The ^{90}Sr leach rate decreased with time and after about 5 years stabilized at the very low rate of 5×10^{-11} g glass/cm^2-day (Merritt 1977).

Recently, development of a stuffed glass process has begun at Catholic University in Washington, D.C. (Simmons et al. 1979). The process utilizes a high-temperature, high-silica glass that can be prepared in a porous form outside the radioactive processing cell. The pre-prepared porous glass is then soaked in HLW solution. After a suitable soaking period the solution-laden porous glass is removed from solution and the HLW constituents are precipitated. The porous glass is then soaked in a solvent that removes the waste from a surface layer of the porous glass. The solvent is subsequently evaporated and the porous glass is dried at 625° to 700°C to convert the HLW constituents in the pores to oxides. Then the temperature is raised to 900°C for sintering. During sintering, the pores collapse. The final product is solid glass that contains the radioactive waste materials interstitially, and has a high-silica envelope on the outer surface. Alternatively, the same final form can be obtained by putting waste-laden porous glass granules in an envelope of waste-free porous glass and sintering to close the pores.

The stuffed glass process potentially yields a product with the durability of a high-melting glass but utilizes lower processing temperatures. In addition, the product has a built-in barrier of inert high silica glass on the surface.

Glass-Ceramics

Glass-ceramics are a class of specially formulated materials that can be melted, processed and formed as glasses and then devitrified, or crystallized, under controlled conditions. Glass-ceramics have become important commercially in the last 20 years. They are valued for their thermal stability and physical ruggedness.

Most of the investigations of glass-ceramics as materials for HLW disposal have been carried out in Germany at the Hahn-Meitner Institute in Berlin and at Karlsruhe (De et al. 1976, Guber et al. 1979). The waste-containing glass-ceramics formulated to date are usually only about 50% crystalline (commercial glass-ceramics are over 95% crystalline). Some improvements in thermal stability (higher softening points) and physical ruggedness have been observed; the leach rates obtained to date are in the same range as those of low-melting waste glasses.

4.3.2.3 Crystalline Waste Forms

For the purposes of this discussion all nonvitreous high-level solid waste forms will be termed crystalline. In general, crystalline waste forms, particularly those that have undergone extensive thermal treatment and are not approaching solid solution limits, are thermodynamically more stable than glass waste forms. In some crystalline waste forms the crystals are "tailored" to resemble minerals that have a demonstrated stability in nature.

Cement

Cements are used routinely to encapsulate low- and intermediate-level radioactive wastes. Liquid or slurry wastes are mixed with a predetermined weight of dry solids. The solids may be primarily Portland cement such as used in concrete, or may consist of cement mixed with fly ash and clays (grouts) and can be specially designed (usually high alumina) cements (Stone 1977 and Lokken 1978).

Cements are intrinsically somewhat porous and due to the hydrated phases are potentially sensitive to damage from radiation and long-term thermal exposure. They have been considered for the treatment of defense HLW, and techniques that reduce the porosity and water content may even make their use for commercial HLW feasible (Roy and Gouda 1978). One such technique is the FUETAP process being investigated at Oak Ridge National Laboratory in which the waste-containing cement mixture is processed at 250°C and 600 psi (Moore et al. 1979).

Calcine

Defense HLW has been calcined using a fluidized bed calcination process at the Idaho Chemical Processing Plant (ICPP) since 1963. Over 1500 m^3 of granular calcined waste particles are now stored in stainless steel bins housed in underground concrete vaults. The calcined waste is a good low-volume, noncorrosive form for storage.

The ICPP calcination process converts the HLW to dry salts and oxides. Consolidation techniques that decrease the surface area of the solids, decrease the potential for airborne fines, and increase the chemical durability are being investigated. The consolidation techniques are either sintering processes that yield a type of glass-ceramic product or processes that embed the pelletized calcine in an inert matrix (INEL 1978, Lamb et al. 1979, see Section 4.3.2.4).

Synthetic Minerals

To create synthetic minerals, nuclear waste constituents are chemically incorporated in crystalline mineral species. The long-term stability of synthetic mineral waste forms can be deduced from the known behavior of analogous naturally occurring minerals. Of course, unavoidable differences, such as radiation effects, must be studied. A review of the stability of minerals that could contain radionuclides is given in Appendix P of Volume 2.

Development of one synthetic mineral concept (called supercalcine) began at Pennsylvania State University and the Pacific Northwest Laboratory (McCarthy 1977, 1979a; McCarthy and Davidson 1975). The concept may be considered an evolution of the well-developed calcination processes. Instead of calcining the liquid HLW as received, additions of calculated quantities of Ca, Al, Si, etc. are made to the HLW so that after calcination and a heat treatment the waste constituents are chemically bound in predetermined mineral assemblages. However, because HLW contains so many different elements, the mineral assemblages tend to be very complex and difficult to characterize. Recently the emphasis in some investigations has switched to the development of stable synthetic minerals for only the actinides in the waste. Fluorite and monazite structures appear to form very stable crystals containing these long-lived waste constituents (McCarthy 1979b). Hot pressing techniques are being investigated for consolidation of the synthetic mineral calcines.

Another synthetic mineral concept being studied extensively is Synroc, an acronym for synthetic rock coined by Dr. A. E. Ringwood of the Australian National University at Canberra, for a concept in which the radionuclides are incorporated in solid solution in just three nonsilicate minerals: hollandite, perovskite and zirconolite (Ringwood et al. 1979). A distinguishing feature of this concept is that it maintains a low waste loading ($\leq$10 wt%) so that the known stability of the host crystals is not perturbed. The waste forms are made by mixing calcined HLW with the Synroc additives and hot pressing at 1200° to 1300°C in sealed nickel containers.

One method of obtaining good accommodation of waste radionuclides in synthetic mineral assemblages is to limit the waste loading, as the Synroc concept does. Conceptually, partitioning the HLW into fractions would simplify the task even further and could permit a higher waste loading. The waste would be partitioned based on considerations of chemical and mineralogical similarities, and the availability of techniques for isolating various waste fractions. The possibility exists of processing each fraction individually into a different synthetic mineral. This concept minimizes crystal compatibility problems during processing and opens up the possibility of using multiple repository sites selected for stability with the various synthetic mineral assemblages made from each fraction.

4.3.2.4 Composite Waste Forms

In composite waste forms, the HLW is usually contained in particles or spheres of one type of material, which is surrounded by one or more different nonradioactive materials. The materials are chosen to have properties that complement one another, so that the properties of the composite are superior to those of the HLW-containing material by itself. The waste-containing material can be particles, spheres, or small pieces of any of the candidate waste forms described in Sections 4.3.2.2 and 4.3.2.3; the surrounding materials are metals or ceramics used to increase thermal conductivity and/or fracture resistance, and possibly to act as additional barriers to the release of radionuclides from the waste-containing core material.

Metal Matrices

The use of metal matrices in composite waste forms has been studied for many years (Lamb 1979, Jardine and Steindler 1978, Neumann 1979). Metal matrices are used to improve thermal conductivity and to minimize fracturing of the waste glass beads by adding ductility, i.e., an ability to bend without breaking, to the composite waste form. A radioactive demonstration of the PAMELA process, in which HLW glass beads are embedded in a lead matrix, is planned as a joint German-Belgium project in the early 1980s (Salander and Zuhlke 1979).

Low-melting metals, such as lead or aluminum and their alloys, have received the most consideration as waste form matrices, but higher-melting metals, such as copper and even steel, can be used to form porous matrices by a powder sintering technique. Even nonporous melt-formed metal matrices may not form a complete barrier to leaching if water contacts the waste form; the bond between the metal and the waste-containing particles may not be tight enough to prevent access of water to the interior of the composite. A barrier can be formed, however, as is done in the PAMELA process, by suspending the waste-containing particles in a basket in the canister and filling the annulus between the basket and the canister wall with pure matrix metal.

Coated Particles

Coated particle composite waste forms are being developed, partially based on the technology developed for the manufacture of high temperature gas-cooled reactor (HTGR) fuels (Rusin et al 1978, 1979a and 1979b). These fuels consist of ceramic pellets that are coated with pyrolytic graphite and silicon carbide, and embedded in a graphite matrix. The core material that has been most studied for coated particle composite waste forms is the synthetic mineral calcine described in Section 4.3.2.3; however, the concept can utilize other core materials. Calcine pellets are formed in a disk pelletizer and coated with pyrolytic graphite and silicon carbide in a fluidized bed. Laboratory tests have shown that an outer coating of durable Al_2O_3 can be added. The coated particles would be surrounded by a metal matrix in canisters before emplacement in a geologic repository.

Coated particles are a way of adapting the multiple barrier concept to the waste form itself. Tests have shown that the particles can have very good chemical durability. However, the processing would be very complex and require a large amount of development before it could be done remotely.

Cermet

This waste form concept, under development at Oak Ridge National Laboratory, produces a uniform dispersion of waste oxide particles within a metal matrix (Quinby 1978). The waste and specific additives required to form the desired ceramic oxide phases and metal alloy matrix are dissolved together in molten urea. The urea solution is precipitated and calcined and the fine powders produced in this step are compacted by extrusion or pressing into desired shapes. In the final processing step the reducible metal oxides, such as oxides of Cr, Ni, Fe, and Co, are reduced in a H_2 or CO atmosphere to form an alloy that encapsulates the unreduced ceramic oxides. After the 800°C reduction the composite is mixed with an organic binder, extruded to form rods and sintered in a nonoxidizing atmosphere at 1200°C to form a dense compact.

High waste loading can be achieved in cermets because metals from salts present in the waste form part of the metal matrix. The reducing conditions reduce volatilization problems during processing.

4.3.2.5 Waste Form Characterization

In that the DOE is committed to examining the relative merits of all potentially available waste forms, research and development is being supported on almost all of the waste forms described in Sections 4.3.2.2, 4.3.2.3 and 4.3.2.4. Treatment processes are already available to produce certain of the waste forms, such as low-melting glass. The DOE program is designed to determine if there are other waste forms that can be practicably produced and that offer improved characteristics. A Materials Characterization

Center has been set up to provide techniques for comparing important waste form materials characteristics on a common basis (Nelson et al. 1980). The first issue of the Nuclear Waste Materials Handbook will be published in approximately two years. It will contain materials data, not only for candidate waste forms, but also for other waste package components.

Since the most likely mechanism for release of radionuclides to the biosphere is reaction with and transport by ground water, resistance to leaching of radionuclides by ground water is the performance characteristic of major interest. Leach resistance can be highly dependent upon the physical, chemical, mechanical, and radiation stability of the waste form. The stability of a waste form depends upon its response to radiation, temperature, and the chemical environment (Mendel et al. 1975). The factors influencing long-term stability are: 1) transmutation by radioactive decay, which may alter the chemical structure of the waste form; 2) recoil from alpha decay, which may break chemical bonds and alter the physical structure of the waste form; 3) heat generated by radioactive decay, which may cause the waste form to change to a more thermodynamically stable state and which may accelerate potential chemical reactions, including leaching; and 4) the chemical environment, i.e., water plus dissolved ions, which ultimately determines the rate of release of radioactive materials into the repository.

4.3.3 TRU Waste Treatment in the Reprocessing Cycle

When spent fuel is reprocessed for uranium and plutonium recycle, the non-high-level and nongaseous wastes that result from these operations and from the mixed oxide fuel fabrication must also be treated and packaged. This section addresses the treatment of these solid and liquid TRU wastes. Treatment and packaging processes for such wastes are described in detail in DOE-ET-0028 (Section 4.0), where wastes are discussed in four categories: 1) fuel residue (the fuel hulls and assembly hardware), 2) failed equipment and noncombustible waste, 3) compactable and combustible waste, and 4) wet and particulte solid wastes. Brief descriptions of the treatment processes for these wastes are given in the following sections; the referenced document may be consulted for details. Both TRU and non-TRU wastes of the latter three categories result from operation of fuel reprocessing plants (FRPs). Only the treatment of the TRU wastes is considered in this Statement; the treatment of the non-TRU portions would be similar, however.

4.3.3.1 Fuel Residue Treatment

Packaging without compaction is the example fuel residue treatment process used in this Statement. Mechanical compaction of hulls and melting of hulls are also described to illustrate other alternatives. The fuel residue packages have surface dose rates well above 0.2 R/hr. Remote handling of these wastes is thus required.

Fuel Residue Packaging Without Compaction (Example Method)

Packaging without compaction is a treatment concept in which the nonsegregated fuel residue is monitored for undissolved fuel, dried, and sealed without compaction in stainless steel canisters (0.76 m dia x 3 m) for shipment to interim storage or to a repository. The void spaces in the canister are filled with dry sand to reduce the possibility of ignition of Zircaloy fines in the fuel residue. Alternatives within the packaging without compaction concept involve separate packaging of the hulls and hardware, deactivation of fines before packaging, and use of filler materials other than sand (e.g., concrete). Other containers (e.g., 55-gallon drums) could also be employed.

Figure 4.3.6, the flow diagram for fuel residue packaging without compaction, shows the steps involved in the process. The quantity of packaged waste resulting from this option is estimated to be 9.1 canisters/GWe-yr.

Mechanical Compaction of Hulls. Mechanical compaction of hulls is a treatment concept for fuel residues in which the hulls are separated from the fuel assembly hardware and Zircaloy fines, compacted to 50% of theoretical density, and packaged in stainless steel canisters (0.76 m dia x 3 m) for shipment to interim storage or to a repository. The Zircaloy fines are deactivated by oxidation and packaged in identical canisters along with the fuel assembly hardware. Compaction of the hulls could be done by a variety of processes, none of which has been evaluated with irradiated hulls. Hydraulic press compaction was selected as the alternative most technically feasible at present.

The steps of the compaction packaging concept are shown in Figure 4.3.7. Implementation of this option is estimated to result in 1.6 canisters/GWe-yr of fuel hardware and 3.8 canisters/GWe-yr of compacted hulls.

Hulls Melting Process

The hulls melting concept considered here uses the Inductoslag melting process developed by the U.S. Bureau of Mines Metallurgical Research Center in Albany, Oregon. In this

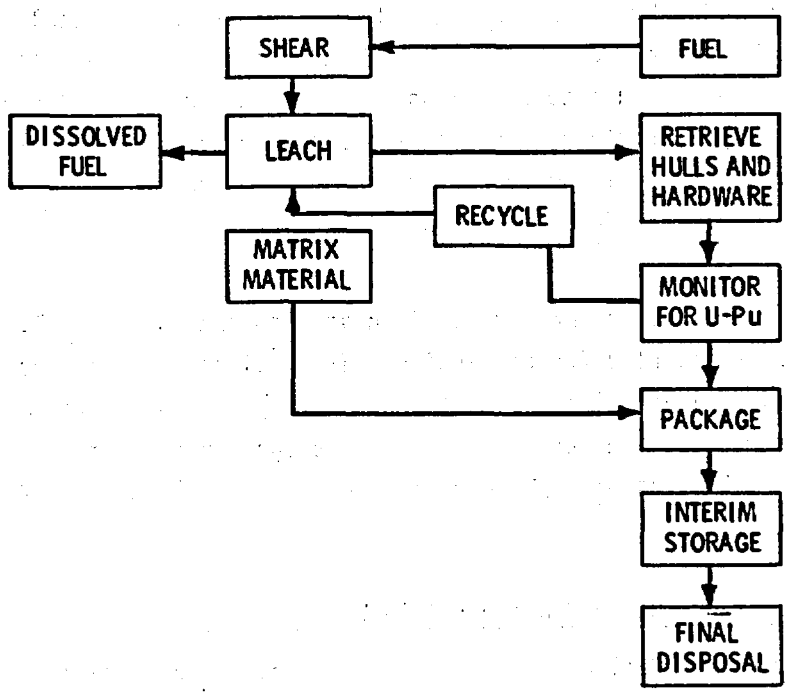

FIGURE 4.3.6. Flow Diagram for Fuel Residue Packaging Without Compaction

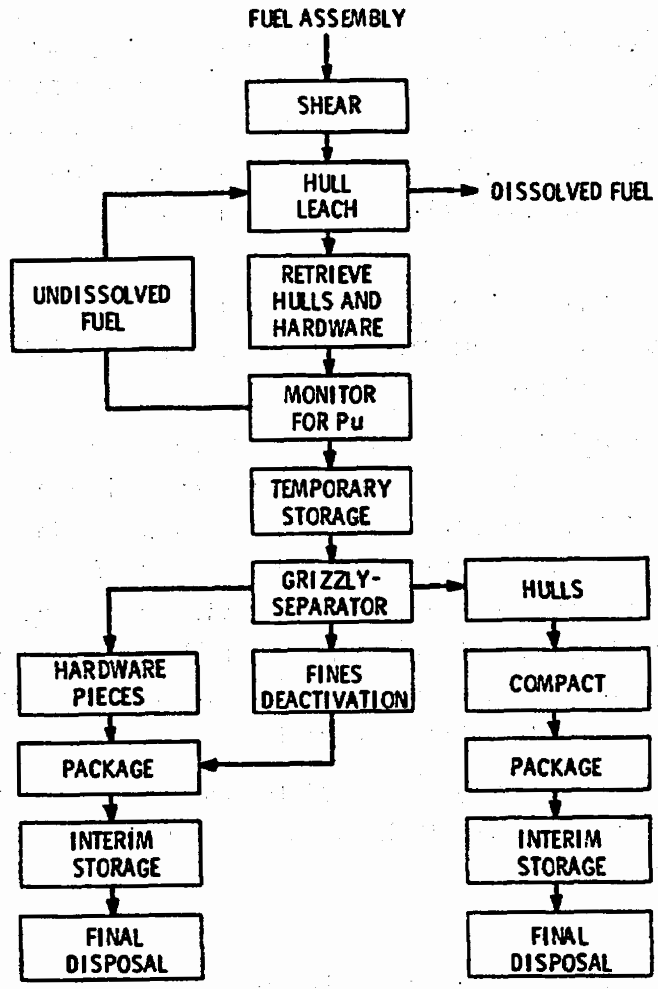

FIGURE 4.3.7. Flow Diagram of Mechanical Compaction of Hulls

process, the sheared cladding hulls are segregated from the stainless steel end fittings and other fuel element hardware and from the Zircaloy fines. The hulls are melted, and the ingots from the melter are sealed into stainless steel containers. The Zircaloy fines are deactivated to eliminate pyrophoric hazards and are packaged with the stainless steel components without melting. This melting concept has been demonstrated successfully in making ingots 10 cm (4 in.) in diameter from simulated fuel residue.

A flow diagram for the melting process is identical to that shown in Figure 4.3.7 except that melting is substituted for compaction. The facility is designed to produce 6 ingots/day, 0.23 m dia x 1.45 m long. These ingots are packaged in 0.76 m dia x 3 m stainless steel canisters, and the fuel hardware is packaged in identical canisters. The estimated quantities are 1.6 canisters/GWe-yr of hardware and 2.1 canisters/GWe-yr of melted hulls.

4.3.3.2 Failed Equipment and Other Noncombustible Waste Treatment

The example treatment of failed equipment and noncombustible waste used in this Statement involves decontamination and disassembly of some of the failed equipment (but not of noncombustible waste), and packaging either in 55-gallon drums, in 1.2 x 1.8 x 1.8 m steel boxes, or (at an FRP) in canisters like those used to contain fuel residue (Section 4.3.3.1). Failed equipment is packaged in canisters when it cannot be decontaminated sufficiently to allow packaging in boxes (the boxes must have a surface dose rate less than 200 mR/hr) or it cannot be disassembled to fit in drums. Figure 4.3.8 is a schematic flow diagram illustrating treatment procedures at an FRP. Procedures at a MOX-FFP are similar in most respects. Alternative treatment concepts involve varying degrees of decontamination and disassembly before packaging and the addition of fixation materials (e.g., cement) within the packages.

For the generic reprocessing cycle studied (Section 3.2.1.2), it is estimated that the quantity of failed equipment resulting from operation of an FRP could be contained in a mixture of packages comprising 1.4 canisters/GWe-yr, 1.1 boxes/GWe-yr, and 9.0 drums/GWe-yr. The boxes have surface dose rates low enough to allow contact-handling but the canisters and drums require remote handling. The noncombustible waste is packaged only in 55-gallon drums; the estimated quantity from an FRP is 84 drums/GWe-yr, approximately 10% of which

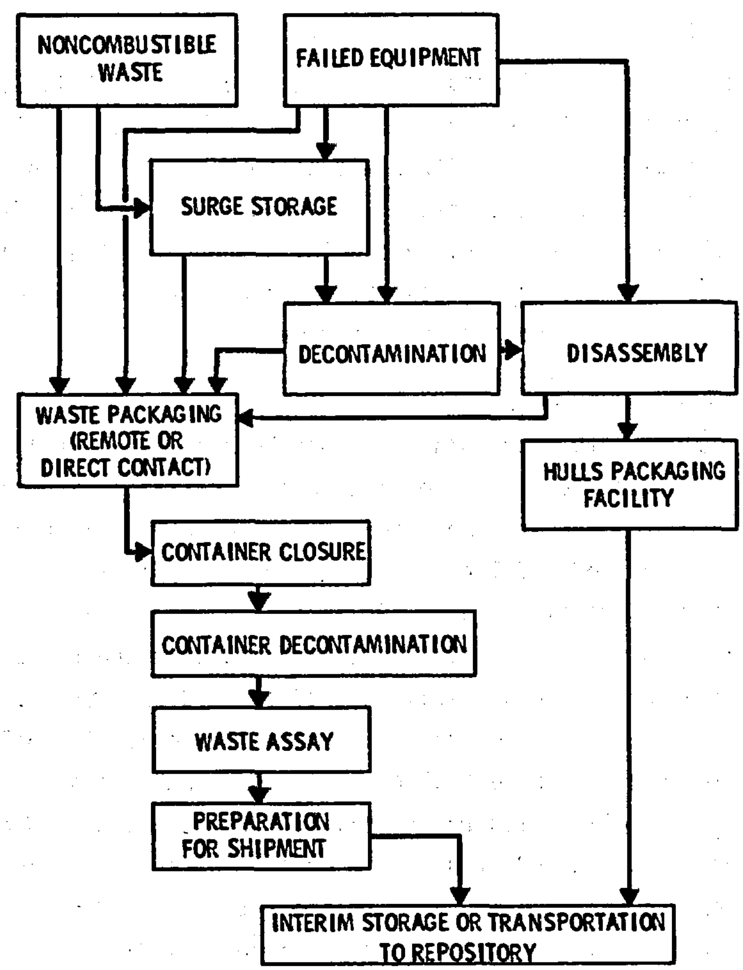

FIGURE 4.3.8. Flow Diagram for Treatment of Failed Equipment and Noncombustible Waste at an FRP

may be contact-handled. The quantities of failed equipment and noncombustible wastes estimated for a MOX-FFP could be contained in a mixture of packages comprising 0.38 boxes/GWe-yr and 7.5 drums/GWe-yr. All of these packages could be contact-handled.

4.3.3.3 Combustible and Compactable Waste Treatment

Three major alternatives have been used for treating general trash and combustible waste: incineration, packaging without treatment, and compaction. Incineration consists of burning the waste and treating the off-gas for removal of radionuclides and other noxious materials, thereby decreasing the waste volume and rendering it noncombustible. Incineration also reduces the potential of biological action occurring in the waste. Packaging without treatment consists of simply packaging general trash and ventilation filters in steel drums for interim storage or interment at the repository. The third alternative, compaction, consists of compacting the waste and packaging it in steel drums for interim storage or interment at the repository. All three methods have been widely used in the nuclear industry, although incineration has not been applied to wastes requiring remote handling. The latter two methods may not give waste packages that meet waste package criteria for the repository.

Incineration was chosen as the example treatment process for this Statement because it both renders the waste noncombustible and reduces the volume. Several incineration processes have been successfully operated with radioactive combustible wastes (Perkins 1976, Borduin and Toboas 1980). The process assumed here and described in DOE/ET-0028 employs a controlled-air, dual-chamber incinerator. Packaging without treatment was also examined in detail as an alternative since it represents the other end of the spectrum in terms of cost, volume reduction, and flammability of the packaged waste.

Incineration (Example Method)

The FRP wastes include both materials that must be handled remotely and those that can be contact-handled; we assume the use of separate but identical incinerators for the two waste categories. The wastes sent to these two units are sorted and high-density combustibles are shredded, as are wooden filter frames after filter media have been removed in a filter media removal and pelletizing press. Pelletized filter media and noncombustibles are packaged in 55-gallon drums for disposal. The sorted and shredded combustibles are incinerated, and the ash (which contains essentially all of the radionuclides present in the waste) is collected for transfer to the wet waste and particulate solids immobilization facility. The off gas from the incinerator is sent through a high-energy gas-scrubbing system for cooling and for removal of volatilized radionuclides, acidic gases, and particulates before being filtered and routed to the FRP atmospheric protection system. The scrubbing solution is concentrated and sent, along with the ash, to the wet waste and particulate solids immobilization facility. Figure 4.3.9 provides a simplified flow diagram of these operations.

We assume that the MOX-FFP is located apart from the FRP and that a separate incineration facility is therefore required. The facility design is nearly identical to that in the FRP; however, because of the relatively small volume of off-gas scrubbing solution, it does not provide for solution concentration before immobilization.

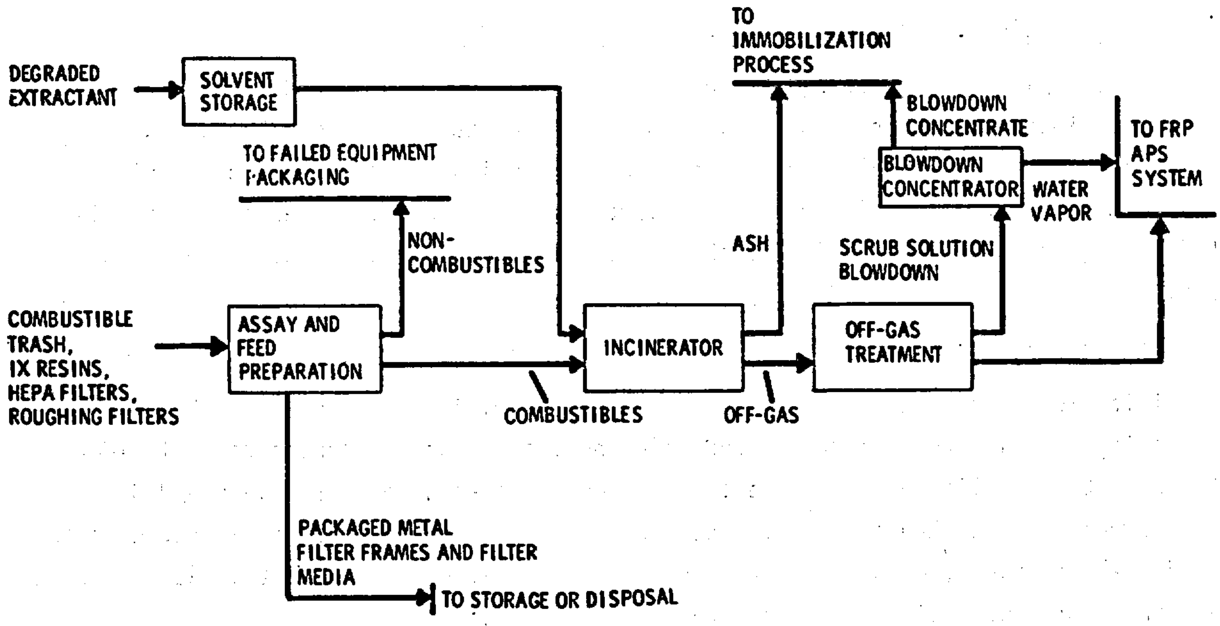

FIGURE 4.3.9. Treatment of Combustible Wastes and Filters at FRP Remotely Handled Waste Incinerator Facility

The only packaged waste outputs from the example incineration facilties are the drums containing the pelletized filter media and minor amounts of noncombustible waste and crushed metallic frames from HEPA filters. The estimated quantities would fill 7.6 55-gallon drums/GWe-yr from FRP operation and 0.95 55-gallon drums/GWe-yr from MOX-FFP operation. The drums from the FRP would require remote handling, but those from the MOX-FFP (because the principle activity results from alpha radiation) could be contact-handled.

Packaging Without Treatment

The waste packages employed for packaging combustible and compactable wastes without treatment are steel drums; the larger HEPA filters are packaged in 80-gallon drums, and the remaining wastes are packaged in 55-gallon drums. The wastes are assumed to be sealed in plastic bags before they are shipped to the packaging facility. In the packaging facility they are examined and placed in new drums (if necessary), assayed for fissile material content, and the lids are tightened to the drums.

The quantities of packaged waste are quite large under this option. We estimate 55 80-gallon drums/GWe-yr and 137 55-gallon drums/GWe-yr of remotely handled waste and 228 55-gallon drums/GWe-yr of contact-handled waste from the FRP. For the MOX-FFP the estimates are 6.6 80-gallon drums/GWe-yr, and 21.5 55-gallon drums/GWe-yr, all of which could be contact-handled.

If the packaging without treatment option is implemented, alternative treatments are employed for two types of combustible waste: ion exchange resins and degraded extractant. The ion exchange resins are sent to the wet waste and particulate solids immobilizaton facility, and the degraded extractant is burned in an incineration unit designed specifically for that purpose.

4.3.3.4 Immobilization of Wet Wastes and Particulate Solids

Prior to shipping and isolating wet wastes, they must be immobilized. This step may be done by a variety of methods. Immobilization of these wet wastes in bitumen and cement (bituminization and cementation) is discussed here as applied to an FRP and a MOX-FFP. Another alternative, urea-formaldehyde immobilization, requires process equipment similar to that for cementation. Cementation is the example treatment process chosen for this Statement.

Cementation (Example Method)

Immobilization of radioactive wet wastes in cement involves mixing the wastes with cement, placing the mixture into drums, and allowing the mixture to harden to a liquid-free product. Cement immobilizaion of radioactive wastes has been widely used in the U.S. A variety of cementation technologies have been developed, including in-drum mixers, drum tumblers, and in-line mixers, each of which is described in ERDA-76-43. For this Statement, a drum-tumbling system was selected for the following reasons:

- Both liquid and dry wastes can be immobilized without altering the commercially available technology.
- The wastes are mixed inside the drums, preventing external solidification of the waste-cement mixture.

The process flow diagram for a cementation system at an FRP is shown in Figure 4.3.10. A similar system can be used at a MOX-FFP after neutralization of the acidic liquids and treatment to remove the ammonia present in those wastes (to avoid possible later pressurization of sealed containers).

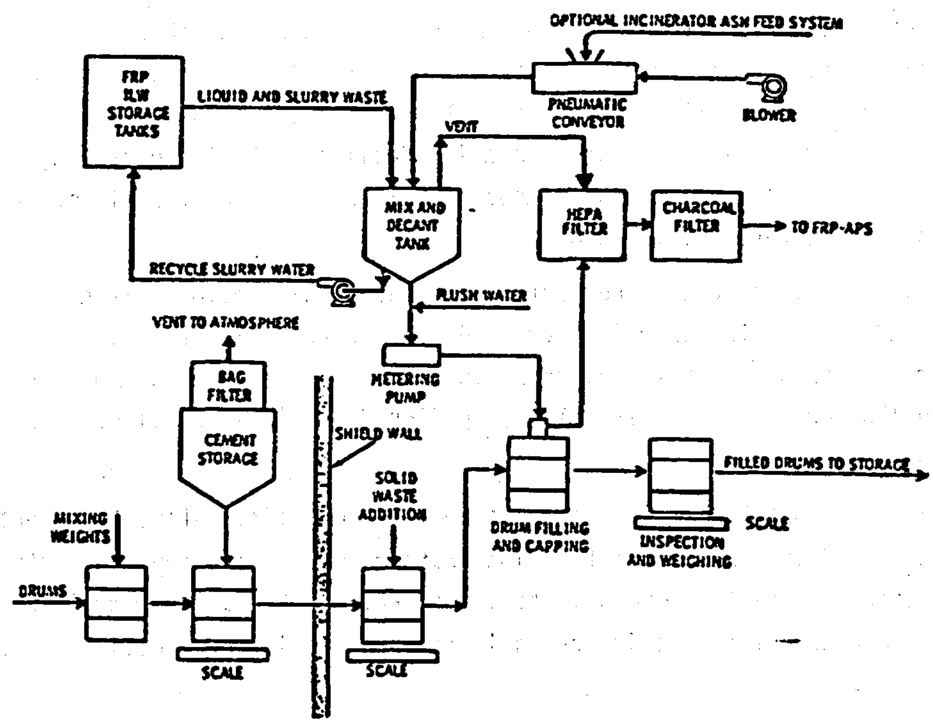

FIGURE 4.3.10. Process Flow Diagram for Cementation at Fuel Reprocessing Plant

The packaged waste output of the cementation systems depends markedly on whether or not the combustible wastes are incinerated (because the incinerator ash and scrubber solutions are additional feeds to the cementation systems). If the combustible wastes are incinerated, the output of the cementation systems will be 106 55-gallon drums/GWe-yr at an FRP and 31 55-gallon drums/GWe-yr at a MOX-FFP. About 40% of the drums originating at an FRP and all of the drums originating at a MOX-FFP could be contact-handled.

If the combustible wastes are not incinerated, the packaged waste output of the cementation systems will be 49 55-gallon drums/GWe-yr at an FRP and 11 55-gallon drums/GWe-yr at a MOX-FFP. All of the drums originating at an FRP require remote handling, but those originating at a MOX-FFP would be contact-handled.

Bitumenization

Immobilization of radioactive wet wastes in bitumen involves mixing the waste with liquid bitumen or asphalt binder and placing it in 55-gallon drums. The temperature of the binder at the time of mixing (above 100°C) evaporates the free water, and thus reduces the waste volume. Use of bitumen to immobilize radioactive wastes has been well demonstrated, largely through extensive operating experience in Europe. However, it is uncertain whether bitumenized waste forms will meet waste form criteria for repositories.

Several types of bitumenization processes have been developed as discussed in ERDA-76-43. In this Statement, a continuous screw extruder process was considered for the following reasons:

- The screw extruder bitumenization process operates at lower temperatures and with shorter residence times than the batch process, thus minimizing off-gas problems.
- The process uses well-demonstrated technology.
- The process is commercially available in the U.S.

A process flow diagram for a bitumenization system at an FRP is shown in Figure 4.3.11. A similar system can be used at a MOX-FFP after neutralization of acidic liquids.

If the combustible wastes are incinerated, the packaged waste output of the bitumenization systems will be 48 55-gallon drums/GWe-yr at an FRP and 10 55-gallon drums/GWe-yr at a MOX-FFP. About 2% of the drums originating at an FRP and all of the drums originating at a MOX-FFP could be contact-handled.

If the combustible wastes are not incinerated, the packaged waste output of the bitumenization systems will be 26 55-gallon drums/GWe-yr at an FRP and 8.7 55-gallon drums/GWe-yr at a MOX-FFP. About 3% of the drums originating at an FRP and all of those originating at a MOX-FFP could be contact-handled.

4.3.4 Gaseous and Airborne Waste Treatment

Spent nuclear fuel contains some radionuclides that are released in gaseous form during certain treatment operations. Such volatile radionuclides include the fission products ^{3}H, ^{85}Kr, and ^{129}I and the activation product ^{14}C. A small portion of the fission product

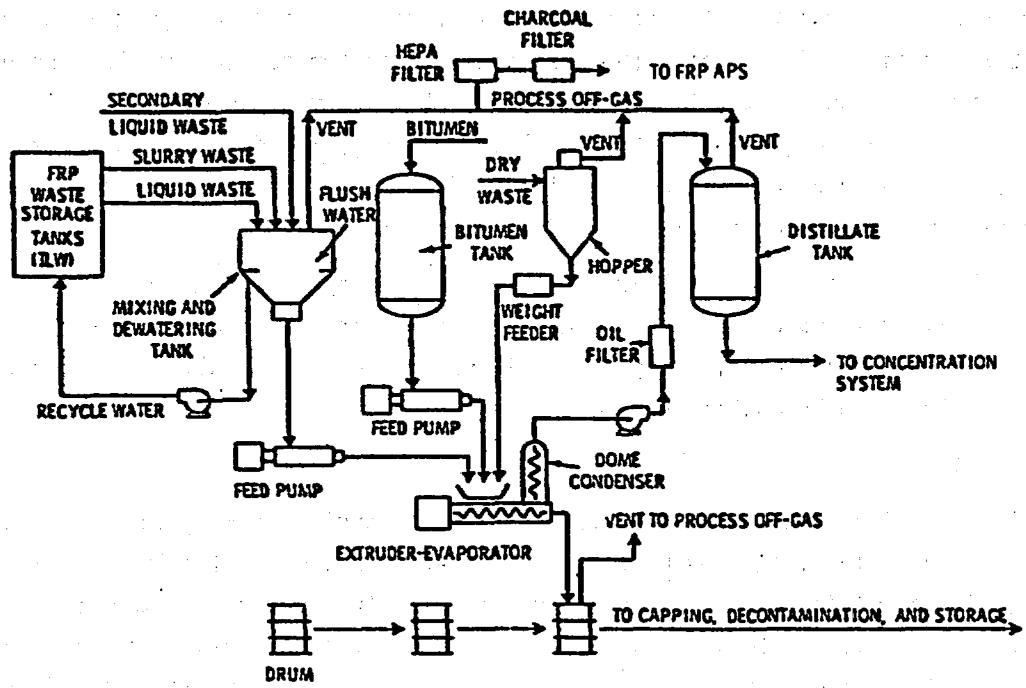

FIGURE 4.3.11. Process Flow Diagram for Bitumenization Facility at Fuel Reprocessing Plant

ruthenium may also be converted to a volatile species under normal process conditions. All of the other radionuclides present may also be present in off-gas and ventilation-air streams; these are present, however, as suspended particles rather than in a gaseous form. The fraction of the nonvolatile radionuclides suspended in the gas streams is generally quite small.

Gaseous and airborne wastes will have to be treated to remove radionuclides whether the spent fuel is discarded (the once-through case) or reprocessed. However, the complexity of treatment operations might vary widely depending on which cycle is chosen. The treatment operations will be at a minimum if spent fuel is packaged as intact assemblies (as in Section 4.3.1.1) and will be at a maximum if spent fuel is dissolved for disposal or reprocessing.

4.3.4.1 Filtration

Filtration is employed to remove radioactive particles from air streams being discharged from various equipment and facilities used in the LWR fuel cycle. Such particles arise from a variety of sources and mechanisms and their release to the environment can be controlled by a variety of filtration processes. There has been much experience in this area, since filtration has been successfully employed for many years in operating nuclear facilities.

One type of filter used almost universally in nuclear installations is the high-efficiency particulate air (HEPA) filter. These filters are composed of a specially formulated glass fiber web contained in a wood or metal frame. HEPA filters are available in several modular sizes; the size most commonly used for large installations is 61 cm on a side by 29 cm deep. Strict quality assurance by the manufacturer and installer ensures that every filter will be at least 99.7% efficient for removing particles of 0.3 µm diameter. A 99.9% efficiency for removing radioactive particles (a decontamination factor (DF) of 10^3) is taken as a reasonable estimate for each stage of HEPA filtration. Higher removals are achieved by the use of multiple stages.

Prefilters are used to remove particles larger than 6 µm and have less efficiency for smaller particles. Prefilters are intended to remove the usual ambient dust from the air stream and thus double or triple the service life of the highly efficient HEPA filter. For radionuclide release calculations, a 91% efficiency for prefilters in removing radioactive particles (a DF of 10) is taken as a reasonable estimate.

Most nuclear facility designs include final filtration of essentially all of the air leaving the facility as well as prior filtration of the air leaving individual portions of the facility (e.g., some process equipment, cells, glove boxes). This is outlined in the flow diagram shown in Figure 4.3.12. The final filtration system has been termed the atmospheric protection system (APS). Three types of atmospheric protection systems are examined in detail in DOE/ET-0028 (Section 4.11) for application at fuel reprocessing plants (similar systems could be used at MOX-FFP and spent fuel treatment facilities). These three APS types use HEPA filters for final filtration but use different types of prefilters. One type of APS employs a commercially available Group III throw-away prefilter, another type employs a sand-bed prefilter, and the third type employs a deep-bed glass fiber filter. The Group III prefilter option was chosen as the example case in this Statement.

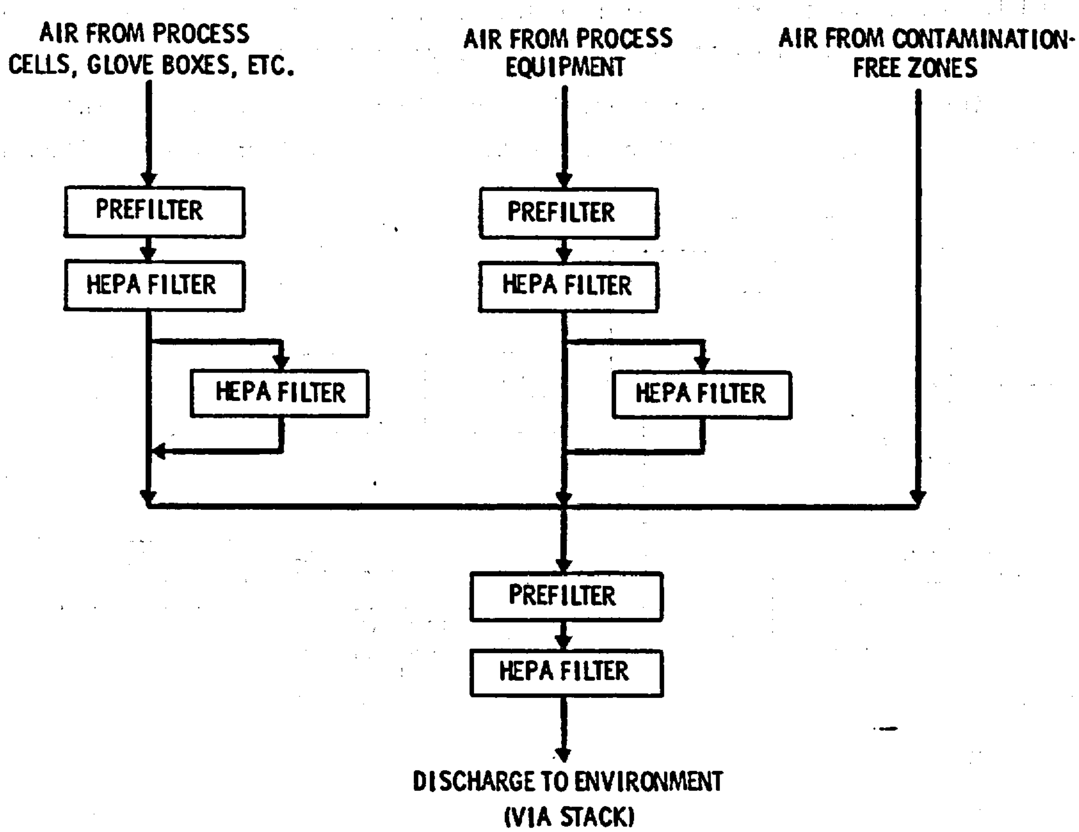

FIGURE 4.3.12. Flow Diagram for Filtration of Airborne Wastes

4.3.4.2 Gaseous Radionuclide Recovery

Where recovery of gaseous radionuclides (i.e., ^{3}H, ^{14}C, ^{85}Kr, ^{129}I) from airborne waste streams is required, processes other than filtration must be employed. Recovery of at least some of these gaseous radionuclides will be required if the spent fuel is processed to convert it to an alternative disposal form in the once-through case or to recover uranium and plutonium for recycle. In the example process of this Statement for the once-through case (the packaging of intact spent fuel assemblies), it is anticipated that no gaseous radionuclide recovery will be required. This is because only small quantities are expected to escape from the fuel.

Recovery of the gaseous radionuclides ^{14}C, ^{85}Kr, and ^{129}I (but not of ^{3}H) is included in the example off-gas treatment process used in this Statement for the reprocessing cycle. Most of this recovery takes place from the off-gas stream leaving the dissolver, since these radionuclides volatilize when the UO_2 fuel is dissolved in nitric acid. Iodine recovery from the gas streams leaving the separations process equipment is also provided, since a significant fraction of the iodine may remain in the dissolver solution and then volatilize later. Figure 4.3.13 presents a flow diagram for this gaseous radionuclide recovery system. The possible use of the voloxidation process to recover tritium is indicated also but, as mentioned previously, tritium recovery is not included in the example process of this Statement.

Tritium (^{3}H) recovery is not included in this Statement because the technology is not believed to have been suitably demonstrated as yet. In the example process, the tritium present in the UO_2 portion of the spent fuel is released to the atmosphere as water vapor. The bulk of this release occurs when the excess water is vaporized and discharged.

Methods of tritium control have been studied. The voloxidation process (Groenier 1977) has received the most development, but other alternatives have also been examined (Burger

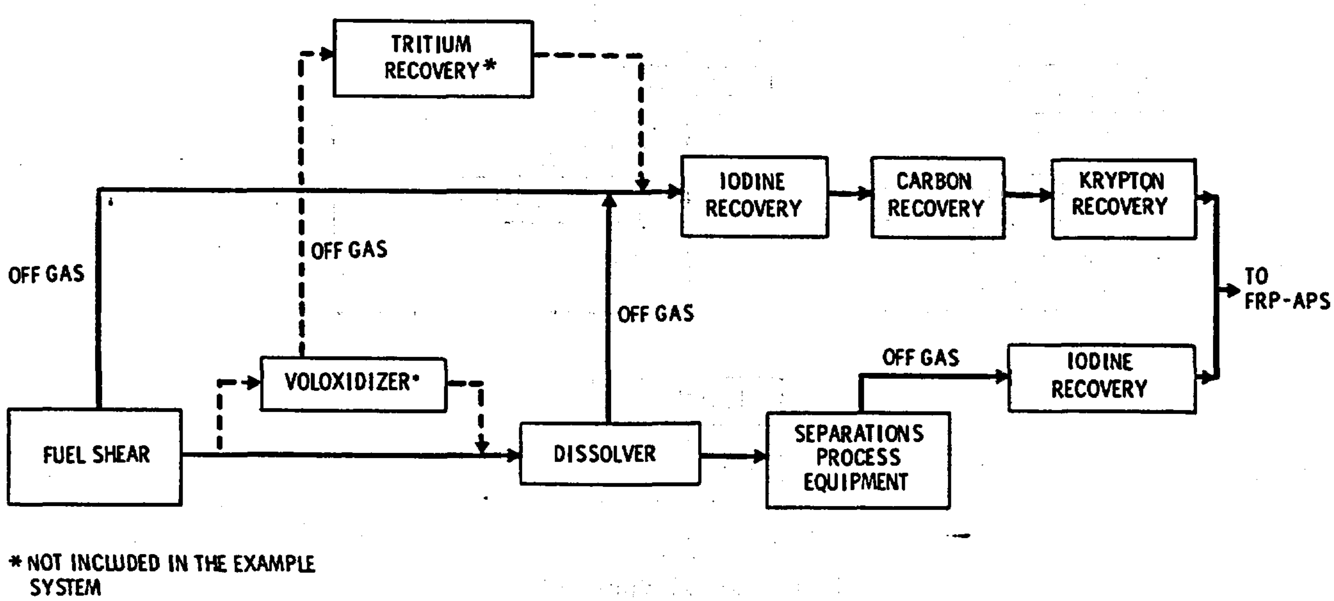

FIGURE 4.3.13. Flow Diagram for Gaseous Radionuclide Recovery

and Scheele 1978). The voloxidation process involves oxidation of UO_2 to U_3O_8 at 400° to 500°C in air. Essentially all of the tritium (plus portions of the other volatile radionuclides) is released to the gas stream by this process. The released tritium is removed from the gas stream (as water) by a bed of adsorbent material.

Although the example process in this Statement includes the recovery of three gaseous radionuclides, the study described in DOE/ET-0028 (Section 4.9) considered other possibilities as well. These included 1) no gaseous radionuclide recovery, 2) recovery of ^{129}I, 3) recovery of ^{129}I plus ^{14}C, and 4) recovery of ^{129}I plus ^{85}Kr.

In the example process, iodine recovery is effected by adsorption on silver zeolite, carbon recovery is accomplished by adsorption (as carbon dioxide) on zeolite molecular sieves, and krypton is recovered by cryogenic (very low temperature) distillation. Silver zeolite is a prepared by replacing sodium ions in a zeolite with silver ions. Zeolite molecular sieves are crystalline aluminosilicates having pores of uniform size that completely exclude molecules which are larger than the pore diameter, thus permitting selective adsorption of those molecules that are smaller than the pore diameter.

The example off-gas treatment system also includes filtration for removal of particulate material, absorption and catalytic destruction steps for the removal of the oxides of nitrogen, NO and NO_2, and ruthenium removal. A small portion of the ruthenium may be converted to a volatile form during processing operations. The example system uses beds of silica gel to remove this ruthenium before it reaches the processes used to recover the gaseous radionuclides.

The ruthenium-loaded silica gel and the iodine-loaded silver zeolite are ultimately disposed of in those forms; the estimated generation rates are 0.046 55-gallon drums/GWe-yr of the ruthenium waste (which requires remote handling) and 0.68 55-gallon drums/GWe-yr of the iodine waste. The carbon dioxide is desorbed from the molecular sieve and converted to solid calcium carbonate for disposal; 0.19 55-gallon drums/GWe-yr is the estimated quantity. The krypton-rich product (80% krypton and 20% xenon) from cryogenic distillation is collected in pressurized gas cylinders for storage; 2.8 cylinders/GWe-yr is the estimated quantity. These gas cylinders will require remote handling.

Alternatives exist for all of the processes employed in the example gaseous radionuclide recovery system. We do not mean to imply that the processes considered here are necessarily the best, only that they are representative of currently available technology. Krypton and carbon could be recovered by fluorocarbon absorption and iodine could be recovered by different solid sorbents or by scrubbing with various aqueous solutions. These alternatives have been discussed elsewhere (ERDA 1976).

4.3.5 Radionuclide Releases During Waste Treatment and Packaging

Estimates have been developed of radionuclide release during waste treatment and packaging operations in both the once-through and the reprocessing cycles. These estimates are summarized in Appendix 10A of DOE/ET-0028 for the packaging of intact spent fuel in a spent fuel packaging facility (SFPF) in the once-through cycle and for a variety of waste treatment options at an FRP and at a MOX-FFP for the reprocessing cycle. Table 4.3.1 contains a summary of the releases estimated for radionuclides of potential importance during the treatment processes selected for use in this Statement. These release estimates are given as the fraction of the quantity present in spent fuel that is released during the treatment and packaging operations.

As mentioned earlier, tritium removal is not assumed in this Statement because the technology has not been fully demonstrated. Should the voloxidation process described earlier be successfully developed and applied, the release of tritium could be reduced to a value only 1% (or less) as large as that listed here.

All of these releases to the environment occur in gaseous or airborne waste streams. There are no planned discharges of radionuclide-contaminated liquid streams from these facilities.

4.3.6 Treated Waste Quantities

Table 4.3.2 contains a summary of the ranges of quantities of treated and packaged high-level, TRU, and gaseous wastes that result from implementation of various options of the once-through or reprocessing cycles described in Sections 4.3.1 through 4.3.4. These quantities are given in terms of the number of waste packages rather than in terms of the volume of waste because, for the mined geologic repository concepts used in this Statement, the repository area required for high-level waste is a function of the waste heat output while the area required for remotely handled TRU wastes is a function of the number of containers rather than of the volume of waste (see Section 5.3). The data for the packaging of intact fuel in the once-through case and for the packaging of the reprocessing wastes were taken from DOE/ET-0028. The data for the packaging of processed spent fuel were taken from ONWI-39.

TABLE 4.3.1. Estimated Radionuclide Releases During Waste Treatment and Packaging

			Release During Treatment and Packaging, Fraction of That in Spent Fuel[a]											
			Fission Products							Actinides			Activation Products	
Cycle	Waste Category	Facility	H	Kr	Sr	Ru	I	Cs	Ce	Pu	Am	Cm	C	Fe, Co, Ni
Once-Through	Spent Fuel	SFPF	2×10^{-6}	6×10^{-5}	1×10^{-12}	1×10^{-12}	2×10^{-5}	4×10^{-11}	1×10^{-12}	0	0	0	6×10^{-6}	1×10^{-10}
Reprocessing	High-Level Liquid Waste	FRP	8×10^{-2}	0	2×10^{-15}	1×10^{-10}	5×10^{-6}	2×10^{-15}	2×10^{-15}	1×10^{-17}	2×10^{-15}	2×10^{-15}	0	0
	Fuel Residue	FRP	6×10^{-7}	0	2×10^{-16}	2×10^{-16}	0	2×10^{-16}	2×10^{-16}	2×10^{-16}	2×10^{-16}	2×10^{-16}	8×10^{-14}	5×10^{-13}
	Failed Equipment and Noncombustible Waste	FRP	2×10^{-20}	0	2×10^{-15}	2×10^{-15}	6×10^{-23}	2×10^{-15}	2×10^{-15}	2×10^{-15}	2×10^{-15}	2×10^{-13}	2×10^{-20}	1×10^{-19}
		MOX-FFP	0	0	0	0	0	0	0	2×10^{-14}	5×10^{-15}[b]	0	0	0
	Combustible Waste and Wet Wastes	FRP	2×10^{-6}	0	6×10^{-17}	3×10^{-15}	2×10^{-3}	6×10^{-17}	6×10^{-17}	3×10^{-14}	3×10^{-16}	3×10^{-16}	5×10^{-17}	3×10^{-16}
		MOX-FFP	0	0	0	0	0	0	0	1×10^{-15}	2×10^{-14}	0	0	0
	Gaseous and Airborne Primary Wastes	FRP	8×10^{-1}	1×10^{-1}	1×10^{-14}	2×10^{-8}	1×10^{-3}	1×10^{-14}	1×10^{-14}	2×10^{-11}	1×10^{-14}	1×10^{-14}	1×10^{-2}	0
		MOX-FFP	0	0	0	0	0	0	0	1×10^{-12}	3×10^{-11}	0	0	0
	Total Wastes from Reprocessing		9×10^{-1}	1×10^{-1}	1×10^{-14}	2×10^{-8}	3×10^{-3}	1×10^{-14}	1×10^{-14}	1×10^{-14}	3×10^{-11}	1×10^{-14}	1×10^{-2}	5×10^{-13}

(a) Quantities present in spent fuel are listed in Tables 4.4.2 and 4.2.4.
(b) Assuming reprocessing 1.5 years after reactor discharge and fuel fabrication one year later.

TABLE 4.3.2. Estimated Quantities of Packaged High-Level, TRU, and Gaseous Wastes

Packaged Waste	Package Type	Packages/GWe-yr: Once-Through Case: Intact Fuel[a]	Once-Through Case: Processed Fuel[b] Low	Once-Through Case: Processed Fuel[b] High	Reprocessing Case: Example	Reprocessing Case: Low	Reprocessing Case: High
High-Level							
Spent fuel	Canister	127	61	141	---	---	---
Solidified Liquid Waste	Canister	---	---	---	35	27[c]	44[c]
Remotely Handled							
Fuel Residue	Canister	---	12	29	9.1	3.7	9.1
Failed Equipment	Canister	---	2	3	1.4	---	---
	Drum	---	---	---	9.0	---	---
Compressed Gas	Canister	---	0.3	0.4	---	---	---
	Gas cylinder	---	---	---	2.8	0	2.8
Other	Canister	---	28	43	---	---	---
	Drum	---	---	---	146	130	316
Contact Handled							
Failed Equipment	Box	---	---	---	1.5	---	---
Other	Drum	---	6.5	9.4	93	29	281
Total		127	110	226	298	190	653

(a) The example case described in Section 4.3.1.1.
(b) For the cases described in Sections 4.3.1.2 and 4.3.1.3.
(c) For canister heat loadings of 1.2 to 3.2 kW, assuming 6.5 years after reactor discharge.

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4.4 WASTE STORAGE

The treated and packaged wastes (Section 4.3) may have to be stored for an interim period of time before they are finally placed in a repository. With some wastes (e.g., spent fuel in the once-through case and high-level waste in the reprocessing cycle case), interim storage is desirable to allow many of the radionuclides to decay; this lowers the rate of heat generation and simplifies the final disposal operations. With other wastes, there is no technical reason for storage prior to disposal, but storage may be required while awaiting availability of a final repository. With yet another type of waste (krypton), a special facility may be required to store the waste until its radioactivity has decayed to a level low enough that it can be released.

4.4.1 Spent Fuel Storage

Storage of spent fuel is an integral part of both the once-through and the reprocessing cycles. In both cases, an initial storage period is aimed at allowing short-lived radionuclides to decay away; this results in a lowered heat generation rate that facilitates subsequent handling operations and also reduces the degree of radionuclide containment required during the processing operations. Unpackaged spent fuel has been stored in water basins in the U.S. for many years. The initial storage period was first envisioned as lasting only about one year, after which the fuel would be reprocessed. However, because of deferral of reprocessing and the possibility that spent fuel may be sent to disposal without reprocessing, and thus require storage until a repository is available, the initial storage period may now last 20 years or more.

Even longer storage before disposal or reprocessing may be desirable or necessary. Thus, extended (up to 100 years) storage of spent fuel has also been examined. Advantages include additional reductions in the radionuclide heat generation rate and the continued availability of the fuel if the decision is made to reprocess spent fuel.

The extended storage concepts examined here involve prior packaging of the fuel, as described in Section 4.3.1.1, although it could well be that water basin storage of unpackaged fuel would be satisfactory for this purpose also. Only intact spent fuel is considered here for extended storage; it is assumed that if spent fuel is to be processed to a different form for disposal, the processing would not be done until the time of disposal. Four storage modes for packaged intact spent fuel are described briefly here along with the water basin storage of unpackaged spent fuel. More detailed descriptions are presented in DOE/ET-0028, Section 5.

Water basin storage is the only method considered in this Statement for unpackaged spent fuel. The four packaged fuel storage concepts are described here to illustrate the range of alternatives available to reduce the already negligible impacts of spent fuel storage to even lower values.

4.4.1.1 Water Basin Storage of Unpackaged Spent Fuel (Example Method)

The storage of spent power reactor fuel in water basins is an established technology that has been used successfully for over 20 years. Water basin storage has been employed at government-owned reactors and commercial light water reactors, fuel storage basins, and a fuel reprocessing plant. The water basin storage of unpackaged spent fuel at independent spent fuel storage facilities and at stand-alone at-reactor basin facilities is discussed in more detail in separate environmental impact statements (DOE/EIS-0015 1980 and NUREG-0575 1979). Water basin storage at independent spent fuel storage facilities was also examined in detail in DOE/ET-0028.

Spent fuel elements arrive at independent storage facilities in shipping casks. The elements are removed from the casks and are placed in storage baskets (containers) that are designed to separate the fuel assemblies sufficiently to assure criticality safety. The baskets are then moved to pool storage positions.

During water basin storage, the pool water serves both as a radiation shield and a heat transfer medium to remove the radionuclide decay heat. This heat is then dissipated to the atmosphere via a cooling tower by means of a secondary (and separate) recirculating cooling system. The water quality in the pool is maintained by filtration and ion exchange.

Two independent water basin storage facilities for unpackaged spent fuel are described in DOE/ET-0028 (Section 5.7). One facility stores LWR fuel assemblies containing 3000 MTHM (metric tons of heavy metal) in six pools (each with a storage capacity of 500 MTHM) and has the capability to receive and/or ship spent fuel at a rate of 1000 MTHM/yr. The other facility is similar but is modified to receive spent fuel at a higher rate and route it to an adjacent fuel packaging facility. This modified facility has the capacity to receive spent fuel at a rate of 2000 MTHM/yr and to store spent fuel containing 3050 MTHM. Other sizes are considered in DOE/EIS-0015.

Radionuclide emissions during operation of such facilities were estimated for receiving and shipping operations and for the storage condition. Table 4.4.1 contains these estimates. These radionuclide emissions occur via the gaseous and airborne release route; no aqueous releases containing radionuclides are expected.

4.4.1.2 Water Basin Storage of Packaged Spent Fuel

The water basin storage of packaged spent fuel is similar to that for unpackaged fuel except that the fuel elements are placed into stainless steel canisters before storage. Packaging of intact spent fuel was discussed in Section 4.3.1.1. These canisters provide additional fuel protection, radionuclide containment barriers, and contamination control.

The facility for water basin storage of packaged spent fuel (see DOE/ET-0028, Section 5.7.5) is somewhat different from that for storage of unpackaged fuel. Each packaged fuel pool is designed to store spent fuel containing 2000 MTHM. The facility is designed for modular expansion to a total of ten such pools for a storage capacity of 20,000 MT.

TABLE 4.4.1. Estimated Radionuclide Releases During Water Basin Storage of Unpackaged Spent Fuel

Fission Products	Fraction[a] Released During Receiving or Shipping	Fraction[a] Released Each Year During Storage
H	2×10^{-6}	1×10^{-6}
Kr	6×10^{-5}	7×10^{-7}
I	1×10^{-7}	9×10^{-9}
Cs	7×10^{-11}	9×10^{-12}
All Others	2×10^{-12}	2×10^{-13}
Actinides	Negligible	Negligible
Activation Products		
C	3×10^{-6}	1×10^{-8}
All Others	2×10^{-10}	2×10^{-11}

(a) Fraction of activity in spent fuel released to atmosphere. See Tables 4.2.2 and 4.2.4 for the activity in spent fuel.

The radionuclide emissions from a facility storing packaged fuel will be markedly lower than those from a facility storing unpackaged fuel. The radionuclide emissions are assumed to be negligible since the containment of the fuel elements in high-integrity packages will reduce the emissions by at least several orders of magnitude below the already low releases resulting from storage of unpackaged fuel.

4.4.1.3 Air-Cooled Vault Storage of Packaged Spent Fuel

Another alternative for extended storage of packaged fuel involves packaging in carbon steel canisters and storing in heavily shielded, air-cooled concrete vaults. The conceptual facility (see DOE/ET-0028, Section 5.7.6), is an adaptation of a storage concept for solidified high-level waste (ARHCO 1976). In this concept natural-draft air circulation is used to remove decay heat so that no mechanical equipment is required for heat removal. The spent fuel canisters are placed vertically within steel sleeves in the vault; these sleeves increase the natural air flow velocity around the canisters and provide additional heat transfer area for the air coolant. Air enters a bottom plenum through side inlets in the structure, passes upward through annuli formed by the storage units and sleeves, and is discharged through an exhaust port to the atmosphere. Air flow is induced by the decay heat of the spent fuel and the design of the vault. This concept has not been used for fuel storage, but is based on established engineering practice and principles.

Double containment of the radionuclides maintains radionuclide emissions at negligible levels. Double containment is provided by single encapsulation of unfailed fuel assemblies (cladding is one barrier and the canister is the second) and by double encapsulation of failed fuel assemblies. A more conservative approach would be to doubly encapsulate all of the assemblies.

The exhaust air is monitored to provide early detection of emissions. If container failure is indicated, the contaminated air is diverted through an adjacent sand filter by forced draft exhaust blowers. The failed package is removed to a facility for repackaging or overpacking. Package failure is expected to be rare or non-existent.

Each sleeve contains either four PWR or nine BWR individually packaged fuel assemblies. The referenced design provides for 1120 sleeves per storage vault and for modular expansion up to a total of ten vaults. Each vault would store spent fuel containing 2000 MTHM, for a total storage capacity of 20,000 MTHM.

4.4.1.4 Dry Well Storage of Packaged Spent Fuel

The concept of dry wells (also called dry caissons) for the storage of packaged spent LWR fuel is similar to concepts already in use for other reactor fuels in both the U.S. (Hammond et al. 1971) and in Canada (Morrisen 1974). For the conceptual facility here (see DOE/ET-0028, Section 5.7.7), the spent fuel is packaged in carbon steel canisters and placed in an underground steel- and concrete-lined caisson. The caisson is then closed with a concrete plug. This concept relies upon the soil to conduct the decay heat from spent fuel to the earth's surface, where it is dissipated to the atmosphere. As in the other packaged fuel storage concepts, double containment is depended on to maintain radionuclide releases at negligible levels.

The caisson is designed so that its atmosphere may be monitored and sampled periodically. Water run-off from the storage area will be collected and monitored (and decontaminated, if necessary) before release. Package failure is considered a highly unlikely event; should it occur, the package is returned to the packaging facility for repackaging or overpacking.

Each caisson provides a storage space of about 1 m in diameter by 5 m high and contains either three PWR or six BWR individually packaged fuel assemblies. The design provides for incremental expansion up to 15,800 caissons, which would store spent fuel containing 20,000 MTHM.

4.4.1.5 Surface Cask Storage of Packaged Spent Fuel

In the surface cask storage concept, packaged spent fuel is stored (outdoors) in a reinforced concrete radiation shield (cask). This concept has been extensively studied (ARHCO 1976) and is a straightforward application of existing technology. In the variation described (see DOE/ET-0028, Section 5.7.8), spent fuel assemblies in carbon steel canisters are placed in vertical concrete casks located outdoors on concrete pads. Heat is removed from the fuel by natural convection air flow upward through the annulus between the cask and the fuel packages.

As in the other packaged fuel storage concepts, double containment limits radionuclide emissions to negligible levels. Monitoring capability is provided to detect radionuclide leakage and also to detect increases in exit air temperature, which would indicate blockage of air ports. Failed packages would be returned to the packaging facility for canister repair or replacement, as necessary; this is considered to be an improbable event.

Each storage unit is about 3.3 m (10 ft) in diameter and about 7.6 m (25 ft) high. Each unit provides a storage envelope of about 1 m in diameter by 5 m high, and contains either four PWR or nine BWR individually packaged fuel assemblies. A large number of storage units would be located at one site; the referenced design provides for incremental expansion up to a total of 11,200 storage units, which would store spent fuel containing 20,000 MTHM.

4.4.2 High-Level Waste Storage

In the reprocessing cycle case where the fuel to be reprocessed has been out of the reactor only a few years, the storage of high-level waste either as a liquid or a solid is desirable to provide additional time for the heat generation rate to decrease. Another potential reason for storage of high-level waste could be to bridge the (possible) gap between waste generation and repository availability. The high-level waste could be stored as a liquid and then be solidified just before repository emplacement, or it could be solidified immediately and then stored in that form until it could be placed in a repository, or it could be stored as a liquid for part of the time and as a solid for part of the time (although the latter case would doubtless be more expensive).

Except for moderate volumes of surge storage in shielded processing facilities, the only method given serious consideration anywhere for interim storage of liquid high-level waste is storage in large underground tanks. Many methods, however, appear suitable for storage of high-level waste after it has been solidified. Solidified high-level waste packages can be stored similarly to spent fuel in water basins, in air cooled vaults, in dry wells, and in casks stored on the surface (ERDA-76-43 1976). Additional details on the storage of liquid high-level waste and of solidified high-level waste in water basins and in sealed casks can be found in DOE/ET-0028 (Section 5).

In the example waste management system considered in this Statement for the reprocessing cycle case, spent fuel is reprocessed 1.5 years after discharge from the reactor. The resultant high-level liquid waste is solidified immediately (except for a minimal surge storage period) and the solidified high-level waste is stored for 5 years in a water basin at the reprocessing plant. When further storage is required pending repository availability, the waste is stored in sealed casks. Certain other waste disposal concepts under consideration (i.e., rock melting and well injection) dispose of high level waste as a liquid. Implementation of one of these concepts may require substantial liquid high-level waste storage facilities.

4.4.2.1 Tank Storage of Liquid High-Level Waste

Storage of liquid high-level waste in large subsurface tanks has been practiced for over 30 years in several countries. Most of the U.S. experience has involved storage of government-produced defense program wastes; the tanks built initially were single-walled, but double-walled tanks have been built in recent years at both Hanford and Savannah River to reduce the possibility of leakage of waste into the environment (DOE/EIS-0063 1980 and DOE/EIS-0062 1980). The defense program wastes were neutralized before storage (by the addition of hydroxides) and are stored in carbon steel tanks. The commercial wastes produced at the West Valley Plant in New York are also stored in this way. More recent plans involve storage of acidic waste in stainless steel tanks. Such tanks have been built (but not used) at the Barnwell Plant in South Carolina. The design concept here (see DOE/ET-0028, Section 5.1) is similar to that used at Barnwell.

The tanks employ double containment, consisting of a primary stainless steel container within a stainless steel liner. Both containers are supported by and encased in a reinforced concrete vault. The tanks in this design are 17 m (54 ft) in diameter and 6 m (20 ft) high and have a net storage volume of 1140 m^3 (300,000 gal) with 10% freeboard. Each such tank has the capacity to store the concentrated high-level liquid waste resulting from reprocessing spent fuel containing 2000 MTHM. Seven tanks are required to provide capacity for 5-yr storage of the high-level waste produced at a 2,000 MT fuel reprocessing plant (four tanks filled, one filling, one emptying and one tank held as a spare). The radioactive decay heat is removed by cooling water, which passes through coils installed in the tanks; the heat is then dissipated via a cooling tower. The contents of the tank are continuously mixed by airlift circulators and by ballast tanks that provide an intermittent flushing action.

The tank off gases are treated to remove any volatilized iodine and particulate radionuclides that might be entrained in the gas stream. Estimated radionuclide emissions are given in Table 4.4.2.

TABLE 4.4.2. Estimated Radionuclide Releases During Tank Storage of Liquid High-Level Waste

Fission Products	Fraction[(a)] Released Each Year During Storage
H	8×10^{-3}
Kr	0
I	5×10^{-7}
Ru	1×10^{-12}
All Others	1×10^{-13}
Actinides	
U	5×10^{-16}
Pu	5×10^{-16}
All Others	1×10^{-13}

(a) Fraction of activity in spent fuel released to atmosphere. See Table 4.2.4 for the activity in spent fuel.

4.4.2.2 Water Basin Storage of Solidified High-Level Waste (Example Method)

Solidified high-level waste packages (described in Section 4.3.2) can be stored in water basins in much the same manner as that described in Section 4.4.1.1 for the water basin storage of spent fuel. In the facility for water basin storage of solidified high-level-waste examined here (see DOE/ET-0028, Section 5.4.1), the singly encapsulated (in stainless steel) waste is received for storage from an adjacent waste solidification facility. The waste canisters are stacked in double-tiered racks in water basins, each of which is designed to hold the waste from reprocessing spent fuel containing 1,500 MTHM. Each basin is equipped with a water purification system and a heat exchanger system to remove the decay heat, which is dissipated to the atmosphere via a cooling tower. Eight such basins are included in the facility design. Radionuclide emissions estimated for water basin storage of vitrified high-level waste are given in Table 4.4.3.

4.4.2.3 Sealed Cask Storage of Solidified High-Level Waste

The sealed storage cask concept for extended storage of solidified high-level waste involves encapsulating the waste canister in a high-integrity, sealed metal storage cask and then placing the doubly encapsulated waste in a reinforced concrete radiation shield. The assembly is then placed on a base in a large outdoor storage yard. Air circulates by natural convection between the radiation shield and the sealed cask to remove the heat being generated by the waste. This concept has been studied extensively (ARHCO 1976).

A facility to implement this concept was designed to accommodate 0.3 x 3 m waste canisters generating about 4.4 kW of decay heat (see DOE/ET-0028, Section 5.4.2). The facility's initial capacity is 2,000 canisters of waste; it can be expanded in 2,000 canister modules to an ultimate capacity of 20,000 canisters.

TABLE 4.4.3. Estimated Radionuclide Releases During Water Basin Storage of Vitrified High-Level Waste

Fission Products	Fraction[a] Released Each Year During Storage
H	0
Kr	0
I	0
Cs	2×10^{-13}
All Others	2×10^{-14}
Actinides	
U	1×10^{-16}
Pu	1×10^{-16}
All Others	2×10^{-14}

(a) Fraction of activity in spent fuel released to atmosphere. See Table 4.2.4 for the activity in spent fuel.

The storage yard is monitored to detect any radionuclide leakage from the storage units. Radionuclide emissions are assumed to be negligible since leakage of the doubly encapsulated waste is believed to be highly improbable. Canisters that do leak can be retrieved and repackaged.

4.4.2.4 Other Solidified High-Level Waste Storage Concepts

Solidified high-level waste could be stored in an air-cooled vault facility similar to that described in Section 4.4.1.3 for the storage of spent fuel. In fact, the conceptual facility for spent fuel storage is an adaptation of a concept for storage of solidified high-level waste (ARHCO 1976). Double containment of the radionuclides in the high-level waste could be provided by overpacking the primary canister. The design for a solidified waste facility would be tailored to the high-level waste canister size and heat generation rate.

Dry well storage of solidified high-level waste could also be employed. This would resemble the dry well storage of spent fuel described in Section 4.4.1.4. Well size and spacing would be different for the solidified waste than for the spent fuel, depending on waste canister size and heat generation rate. Double containment of the waste by overpacking the primary canister could also be utilized for this storage concept.

4.4.3 TRU Waste Storage

The packages of treated TRU waste described in Section 4.3.1 for the once-through case and in Section 4.3.3 for the reprocessing case could require storage for an interim period before a repository is available.

The packaged wastes are considered in one of two categories depending on the radiation level. Packages that have surface dose rates no higher than 200 millirem/hr are "contact-handled," i.e., workers can handle them without extensive shielding. Packages with higher surface dose rates require shielding and/or remote handling to protect operating personnel; these packages are "remotely handled."

The TRU waste packages with the highest surface dose rates are the canisters containing the fuel residues (the fuel hulls and hardware). Some disassembled failed equipment is also assumed to be packaged in identical canisters. Two alternative interim-storage facility concepts for these canisters are described here (see also DOE/ET-0028, Section 5.2): vault storage and dry-well (near-surface) storage. The dry well concept is used as the example method in this Statement.

Other remotely handled TRU wastes are packaged in steel 55-gal drums. Vault storage and dry well storage facility concepts for these wastes are described here (see also DOE/ET-0028, Section 5.3). Vault storage is used as the example method in this Statement.

The contact-handled wastes are packaged in steel boxes or drums. Unshielded indoor storage and outdoor surface storage facility concepts are described for these wastes. The outdoor surface storage concept is the example concept used in this Statement.

Because of the lower radionuclide content and the integrity of the waste packages, no significant releases of radionuclides are anticipated from any of these conceptual TRU waste storage facilities. However, effluents would be monitored to verify that this is indeed true and to provide early detection of problems that might arise.

4.4.3.1 Vault Storage of RH-TRU (Example Method for Drummed RH-TRU)

In the vault storage concept for remotely handled wastes, the waste is considered to be packaged either in special canisters (0.76 m dia x 3 m) or in 55-gal drums. Vault storage is the example concept of this Statement for these 55-gal-drum-packaged wastes and an alternative concept for these canistered wastes.

The 55-gal drums that require remote handling are simply stacked in cells constructed of reinforced concrete. The drums are unloaded from the shipping container and are placed in the storage cells by a crane using a vacuum-operated lifting device. The design calls for each cell to contain 500 drums; these are five layers of drums, 100 drums in each layer, and plywood sheets separate the layers. The basic storage module contains 40 such cells holding a total of 20,000 drums. Facility designs were evaluated for storage both at an individual fuel reprocessing plant and at an independent site serving a number of reprocessing plants.

The vault storage concept for the canistered waste uses individual sleeves for canister storage in concrete vaults, which provide radiation shielding. The canisters are handled with a remotely operated crane. They are lowered from shipping casks through a special transfer device into the storage space and a shielding plug is placed above the canister. Each storage space is a galvanized steel pipe (0.9 m in dia) with a plate welded to the bottom and is suspended from the roof slab of the vault. Natural air circulation through the vault provides canister cooling. The vault storage concept for canisters is based on a modular design. Each cell has a capacity of 312 canisters. Facility designs were evaluated for siting both at an individual fuel reprocessing plant and at an independent site serving a number of reprocessing plants.

4.4.3.2 Dry-Well Storage of RH-TRU (Example Method for Canistered RH-TRU)

The dry-well storage concept, which is the example concept of this Statement for the storage of canisters containing the fuel residue and some of the failed equipment, involves construction of storage spaces in an above-grade soil structure (berm). The canisters are placed in individual storage spaces positioned vertically in the berm, and the spaces are capped with steel and concrete plugs. The plug, canister, and shipping cask are handled remotely using a crane. Each storage space consists of a galvanized steel pipe sleeve (0.9 m in dia) with a plate welded to its bottom and suspended from a slab; gravel is backfilled around the outside of the pipe. Heat is removed by conduction through the soil to the atmosphere. The basic module designed for the dry-well storage of canisters has two berms, each containing 1,248 storage spaces.

A similar approach was examined as an alternative for the storage of the waste packaged in 55-gal drums that requires remote handling. In this instance 5 drums are stored in each caisson (0.66 m dia x 5.2 m deep). Most of the drums can be unloaded from the shipping container and placed in storage using only a shielded mobile yard crane that has a vacuum lifting device. Drums having high surface dose rates are transferred to the caisson using a bottom loading cask. In this design, 504 storage spaces are provided in each module.

4.4.3.3 Unshielded Indoor Storage of CH-TRU

The packages of TRU waste that can be contact-handled can be stored indoors in an unshielded facility. A conceptual facility examined as an alternative to outdoor storage consists of a precast concrete building containing a number of individual storage cells. Drums (55-gal) are stacked six high in horizontal layers; plywood sheets are placed between the layers. Steel boxes are also used to package such wastes; a storage box occupies the space of 12 drums. The boxes and drums are handled by mobile cranes and by fork-lift trucks.

The basic module used in this design includes two cells, each of which will store 4,200 drums. When storage capacity beyond that provided by the basic module is required, an expanded version of the basic module is used or multiples of the basic module are employed.

4.4.3.4 Outdoor Storage of CH-TRU (Example Method)

Outdoor storage is the example concept of this Statement for contact-handled TRU wastes. This approach is presently used at most government installations. Several variations are in use, involving below-grade as well as above-grade techniques and differing amounts of weather protection. The most widely accepted method is to place the waste packages on some structural pad, and cover them first with an impermeable membrane, and then with dirt.

In this design the drums and boxes of waste are placed on an above-ground asphalt slab that is contained within a temporary air-supported structure to allow operations to continue during inclement weather. The containers are arranged in horizontal layers; sheets of plywood are placed over each layer before the next layer is added. Handling of the containers is by mobile crane and by a drum grabber. As the storage area is filled, polyethylene sheets are placed over the stacked containers and the stack is covered with dirt to a depth of at least 0.9 m. Once a storage area is completely filled and covered with earth, the air-supported structure is removed, and the dirt cover is either seeded or covered with a bitumen layer.

The basic storage module for this concept has a storage capacity for 10,000 55-gal drums of waste. Capacity can be expanded by either using an expanded version of the basic module or by using multiples of the module.

4.4.4 Krypton Storage

The ^{85}Kr removed from the off-gas stream as described in Section 4.3.4.2 must also be stored. This gaseous radionuclide can be encapsulated and stored in pressurized gas cylinders. Alternative krypton encapsulation techniques being investigated include 1) zeolite encapsulation, where krypton is diffused into "crystalline cages" at high temperatures and pressures, and where escape of the krypton is slow at low temperatures; 2) dissolution in a glass matrix, where krypton is trapped within a glass when it solidifies; and 3) entrapment of krypton in metal solids during high-rate sputtering.

The krypton storage facility chosen for this Statement stores gas cylinders containing about 80% krypton and 20% xenon. The radionuclide heat generation rate from such cylinders is appreciable and refrigerated air cooling is provided. The surface dose rates of the cylinders are such that remote handling is required; this is provided by special transfer containers and cranes.

The storage plan for krypton differs from those for\the other wastes in an important respect. Since the half-life of ^{85}Kr is relatively short (10.7 yr), it is assumed that after storage for 50 years or so the ^{85}Kr can be released. In 50 years the amount of ^{85}Kr remaining will be only 4% of the initial amount; after 60 years only 2% will remain.

The krypton storage facility (see DOE/ET-0028, Section 5.6) is located adjacent to a fuel reprocessing plant and is sized to handle the output of the plant during its lifetime. Separate storage cells, each holding 104 cylinders, are provided. The number of cells is increased every ten years to provide the necessary storage capacity; 14 cells are required for each ten years' output. The facility also includes hot cells for use in cylinder inspection and gas transfer (e.g., from a leaking cylinder to a sound cylinder) operations.

The gas cylinders are passed into the storage cell through ball valves and rest horizontally on shelves within the cell. Each storage cell contains five shelves and is provided with a self-contained air circulation and heat removal system. These air circulation systems are monitored to provide detection of leaks. If a minor leak is detected, the cylinder is sent to the hot cell and the contents are transferred to a new cylinder. If a cylinder suddenly ruptures, the cell atmosphere will be pumped to a holding tank where it will be sampled and then either returned to the fuel reprocessing plant or sent to the storage facility stack for release.

The normal release of ^{85}Kr from the storage facility occurs in two ways: 1) the small leakages from a number of cylinders, and 2) the planned discharge of the krypton at the completion of the storage period. The former release is estimated to amount each year to no more than 0.1% of the amount of ^{85}Kr present during the year. The latter release does not begin until completion of the planned storage period. For a 50-yr storage period, this release amounts to 4% of the amount initially placed into storage. The planned storage period (and, thereby, the planned release) can be changed after storage has begun.

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4.5 WASTE TRANSPORT

For the example once-through cycle, the waste transportation of concern for this Statement is the shipment of spent fuel. Other wastes would be non-TRU wastes that are not covered in this Statement. The spent fuel may be shipped directly from the nuclear power plants to an encapsulation facility located at the geologic repository site, or it may be shipped first to an interim storage facility and then to the encapsulation facility.

For the reprocessing cycle, transportation is considered for spent fuel, solidified high-level waste, and TRU wastes. Spent fuel may be shipped from the reactors either to interim storage or directly to reprocessing. Reprocessing plant and MOX fabrication plant waste packages may be shipped directly from the fuel reprocessing plants and from the mixed oxide fuel fabrication plants to the geologic repository, or they may be shipped first to an interim storage facility and then to the geologic repository.

The transportation of these wastes is discussed briefly in the following sections. More detail is contained in Section 6 of DOE/ET-0028.

4.5.1 Spent Fuel Transport

Spent fuel has been shipped in the United States for many years. Massive, heavily shielded shipping casks are available for both truck and rail transport of spent fuel from current-generation LWRs. Most spent fuel casks will accept either PWR or BWR spent fuel by using different fuel baskets; however, some are designed only for a particular fuel type. Table 4.5.1 gives information about casks that are currently available or licensed for spent fuel shipments in the U.S. More detailed information is contained in Sections 6.2.1 and 6.2.2 of DOE/ET-0028 and in Volume 2, Appendix C of DOE/EIS-0015.

TABLE 4.5.1 Available Shipping Casks for Current Generation LWR Spent Fuel

Cask Designation	Number of Assemblies PWR	Number of Assemblies BWR	Approximate Loaded Cask Weight, MT	Usual Transport Mode	Shielding Gamma	Shielding Neutron	Cavity Coolant	Maximum Heat Removal, kW	Number Available[a]
NFS-4 (NAC-1)	1	2	23	Truck	Lead and steel	Borated water and antifreeze	Water	12	7
NLI 1/2	1	2	22	Truck	Lead, uranium and steel	Water	Helium	11	5
TN-8	3		36	Truck[b]	Lead and steel	Borated solid resin	Air	36	2
TN-9		7	36	Truck[b]	Lead and steel	Borated solid resin	Air	25	1
IF-300	7	18	63	Rail[c]	Uranium and steel	Water and antifreeze	Water	76[d]	4
NLI 10/24	10	24	88	Rail	Lead and steel	Water	Helium	97[e]	2

(a) According to Winsor, Faletti, and De Steese (1980).
(b) Overweight permit required.
(c) Truck shipment for short distances with overweight permit.
(d) Licensed decay heat load is 62 kW.
(e) Licensed decay heat load is 70 kW.

These existing casks were designed to transport short-cooled (6 months or less) irradiated fuel, consistent with the earlier expectation of rapid recycling of fissile materials. The current situation, however, indicates that most spent fuel transport will involve fuel that has been cooled for at least several years. Consequently, there appears to be considerable incentive to build a fleet of casks specifically designed for this long-cooled fuel because its lower thermal and radiation output would permit an increase in cask capacity and a reduction in handling costs. Several cask fabricators have announced new cask construction programs; some of these address the prospect of transporting long-cooled fuel.

Existing cask designs are for the transportation of unpackaged spent fuel. Transportation of spent fuel that has been packaged in canisters (either as intact spent fuel or as treated spent fuel) will require some additional design modifications. If existing casks or cask designs cannot be suitably modified, new cask designs may be required.

Past experience indicates that an estimated six to eight years could be required to design, test, license, and then fabricate a fleet of newly designed casks. However, with a licensed standard cask, a vendor could significantly shorten the length of time required to deliver a fleet of casks. The useful life of spent fuel shipping casks is estimated to be 20 to 30 years.

Several factors can influence the choice of rail or truck casks for use in the shipment of spent fuel. Rail casks have a significantly larger payload than truck casks. About 10 times as much fuel can be shipped in a rail cask with an increase in shielding weight of only about a factor of 4 over the amount required for a truck cask. On the other hand, truck shipments normally require less time for completion than rail shipments. About 50% of the reactors now operating in the U.S. or scheduled for completion by 1980 do not have rail spurs at the site. Many of these reactors without rail spurs can be serviced by intermodal (truck or rail) casks, which require overweight permits for shipment by truck to the nearest rail siding.

In this Statement, it is assumed that 90% of unpackaged spent fuel will be shipped from reactors by rail and 10% by truck. To accommodate the reactors without rail access, half of the rail shipments are assumed to be in intermodal casks that allow truck shipment for short distances. Shipments from interim storage to repositories or reprocessing are assumed to be 100% by rail. Any shipments of packaged spent fuel are assumed to be by rail using casks that can handle 7 PWR or 17 BWR packaged assemblies. Spent fuel in the once-through cycle is assumed to cool at least five years before shipment. In the assumed reprocessing cycle, however, spent fuel (which is not a waste in this cycle) can be shipped to a reprocessing plant after one year cooling.

Transport of spent fuel by barge and by ship has also been considered. Barge transport is an alternative when both the nuclear power plant and the encapsulation or storage facility are on navigable waterways. Barge transport suggests high payloads and low tariffs. However, cost gains in these two areas could be offset by the longer transit times estimated for barge shipments. Should offshore (floating) nuclear power plants be constructed, barge transport is an obvious choice for the initial portion of the journey of the spent fuel to an encapsulation or storage facility. Casks for barge shipment of spent fuel would probably be similar, if not identical, to those used for rail transport.

Ship transport of spent fuel could be required if some of the alternatives to geologic disposal (e.g., island, subseabed, icesheet) described in Chapter 6 of this Statement are implemented. Casks for spent fuel transport by ship would probably require adaptation or modification of existing design. The design would likely vary somewhat depending on the specific disposal concept, but could be similar to those of existing casks.

4.5.2 High-Level Waste Transport

High-level waste transport is required in the example reprocessing cycle. Solidified high-level waste could be shipped in specially designed casks by truck, rail, barge, or ship, much the same as for spent fuel. Ship transport would be employed only if a disposal alternative involving transport across an ocean were implemented. Barge transport would likely be employed only if both the repository and the fuel reprocessing plant were located on or very near navigable waterways. Rail transport would likely be preferred to truck transport because of the greater capacity of the rail casks.

We assume in this Statement that all transport of solidified high-level waste is by rail. Casks for such use have not been constructed but some have been designed (Perona and Blomeke 1972, Peterson and Rhoads 1977). These designs provide for transport of multiple waste canisters in a single cask and incorporate many features of spent fuel cask designs.

The rail cask chosen as the basis for this study is a lead-filled double-walled stainless steel cylinder weighing about 100 MT (220,000 lb) (Peterson and Rhoads 1972). Neutron shielding is furnished by a water jacket that surrounds the cask body. The cask will dissipate up to 50 kW of internally generated heat. High-level waste canisters are held in an aluminum insert that fits into the cask cavity. Different inserts can accommodate nine 0.30-m dia (12-in.), thirteen 0.25-m dia (10-in.), twenty 0.20-m dia, or thirty-six 0.15-m dia (6-in.) waste canisters. Each of these configurations transports the same quantity of waste. Thus, regardless of the canister heat generation limit imposed by disposal constraints, the required number of shipments does not vary.

The cask is transported on a special six-axle rail car. The gross shipping weight of the loaded cask and rail car is about 350 MT (330,000 lb). Casks used for ship transport, in the event this is required by the choice of a disposal alternative, would require adaptation or modification of existing design.

4.5.3 TRU Waste Transport

Transport of TRU wastes is also required in the reprocessing cycle. These wastes are considered here in two categories: 1) fuel residues, which we assume to be packaged in special canisters; and 2) other solid wastes, which we assume to be packaged in steel drums or boxes (except for a small quantity in special canisters). Only truck and rail transport are considered.

4.5.3.1 Fuel Residue Transport

Fuel residues (spent fuel hulls and hardware) are assumed in this Statement to be packaged in special stainless steel canisters (Section 4.3.3.1). Casks for transport of such canisters have not been built, but it is reasonable to assume that the design and construction of such casks present no new problems.

Fuel residue casks may be shipped by rail or truck. Because rail casks could have a greater capacity and because both reprocessing plants and repositories will have rail service, we assume in this Statement that all fuel residue shipments are by rail. For planning purposes a rail cask has been postulated that would transport three canisters. The conceptual cask is a lead-filled, double-walled stainless steel cylinder weighing about 45 MT (140,000 lb). An insert would position the three canisters inside the cask cavity and would act as a heat conduction path from the waste canisters to the inner surface of the cavity wall. Neither cooling fins nor neutron shielding are required.

A truck cask that would transport one fuel residue canister has also been postulated for comparison purposes. This conceptual truck cask is assumed to be a lead-filled, double-walled stainless steel cylinder weighing about 20 MT (43,000 lb).

4.5.3.2 Other TRU Waste Transport

Other TRU wastes to be transported are the packages resulting from the treatment and packaging operations for failed equipment and other miscellaneous TRU wastes (described in Sections 4.3.3.2 through 4.3.3.4). These packages are mainly steel drums and steel boxes, but special canisters like those used for fuel residue are used in this Statement for a portion of the failed equipment. We assume that all of these packages require shipment in casks or overpacks that meet Type B packaging standards, even though it is likely that some could contain a small enough quantity of radioactivity to permit their shipment in Type A packages. Typical Type A packaging includes steel drums, wooden boxes, and steel boxes that prevent loss or dispersal of radioactive contents and retain radioactive shielding if required when subjected to stresses associated with normal transport. Type B packaging must meet these standards, but also must be able to survive a series of hypothetical accident test conditions.

Shipments of these wastes could be made by truck or rail. We assume here that most of these shipments will be by truck. The special canisters containing some of the failed equipment are transported by rail along with the fuel residue waste.

Drums and boxes that have surface dose rates below 200 mR/hr and can be contact-handled are assumed to be transported in a Super Tiger.® A Super Tiger is a double-walled steel box with a fire-resistant polyurethane foam filler for shock and thermal insulation. Three pallets, each containing twelve 55-gal drums or three steel boxes (1.2 x 1.2 x 1.8 m), can be accommodated in a Super Tiger. The maximum payload is about 14 MT (30,000 lb), and the empty weight is 6.8 MT (15,000 lb). Super Tigers can be carried by either truck or rail.

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Drums that have surface dose rates in the range 200 mR/hr to 1 R/hr require remote handling and are assumed here to be transported in a shielded van that meets Type B package standards or in a Super Tiger-type overpack that incorporates some shielding even though such packages are not currently available or designed. Drums that have surface dose rates in the range 1 to 10 R/hr are assumed here to be transported in casks having an equivalent shield thickness of 5 cm lead + 2 cm steel; a capacity of 14 drums per cask is assumed. Drums with surface dose rates above 10 R/hr are assumed to be transported in casks with an equivalent shield thickness of 10 cm lead + 2.5 cm steel; a capacity of six drums per cask is assumed for planning purposes.

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4.6 DECOMMISSIONING OF RETIRED FACILITIES

Portions of fuel cycle facilities become contaminated with radionuclides during their use. Upon retirement these facilities become a waste that must be managed. Management of this waste is commonly termed decommissioning. Various alternatives are available for decommissioning retired fuel cycle facilities, as discussed in DOE/ET-0028, Section 8.0. Much of this information was extrapolated from results of detailed studies of the technology, safety, and costs of decommissioning nuclear facilities that have been performed at PNL for the NRC (see Schneider and Jenkins 1977, Smith et al. 1978, Smith and Polentz 1978, Jenkins et al. 1979). In this Statement we assume that dismantlement is required and have chosen one of two basic decommissioning modes: either immediate dismantlement, or safe storage with deferred dismantlement.

In immediate dismantlement, all radioactive contamination above regulatory limits is removed from the facility to an approved disposal or storage site shortly after the facility is shut down. Depending on further uses of the site, noncontaminated portions of the facility remaining after dismantlement may be demolished and removed or they may be used for other purposes.

In safe storage with deferred dismantlement, the facility is prepared at shutdown to be left in place for an extended time before it is dismantled. The purpose of this deferment is to allow some of the radionuclides to decay so that radiation exposure during the decommissioning will be reduced. Consideration has been given to both passive safe storage and hardened safe storage methods. These methods differ in the strength and complexity of the barriers installed and in the amount of maintenance and surveillance required during the time of deferment. This time period is termed the continuing care period.

Among the techniques used in decommissioning are chemical decontamination, mechanical decontamination, equipment deactivation and removal, and isolation of contaminated areas. Chemical decontamination is often carried out during the initial stages of a decommissioning operation to reduce radiation levels and remove relatively mobile contamination. Decontamination solutions may include corrosive acids, complexants, detergents, and high-pressure water or steam. These liquids are generally concentrated by evaporation, and the concentrated waste is then immobilized for disposal or storage.

Mechanical decontamination is required to remove residual radioactive contamination from structural surfaces. These activities are minimal when the facility is being prepared for safe storage but are extensive during dismantlement. Contaminated steel structural components or liners may be removed by sectioning in place with plasma torches, arc saws, or explosives. Contaminated concrete can be removed with explosives, by drilling and rock-splitting, or by jackhammering.

Equipment deactivation is done during preparation for safe storage and equipment is removed at the time of dismantlement. Deactivation involves removing bulk quantities of process materials or other hazardous substances, closing valves or installing blank flanges, and disconnecting electricity and other utilities. Steel equipment can be sectioned (if necessary) and removed using cutting torches, saws, and/or explosive cutting techniques.

Isolation of contaminated areas is required for safe storage. Airtight barriers are constructed around contaminated areas (existing facility structures form most of the barrier) and existing penetrations into contaminated areas are sealed off. HEPA-filtered vents may be installed to accommodate changes in air pressure caused by temperature fluctuations. The barriers constructed for hardened safe storage typically are more substantial and require less maintenance during the continuing care period than the barriers constructed for passive safe storage.

This Statement addresses decommissioning only of the fuel cycle facilities subsequent to the nuclear power plants and decommissioning waste treatment of only the TRU wastes. All of the decommissioning wastes from the example once-through fuel cycle and a portion of those from the reprocessing fuel cycle are expected to be non-TRU wastes.

The fuel cycle facilities examined in detail in this Statement include the away-from-reactor storage facilities (AFRs) in the once-through cycle and fuel reprocessing plants (FRPs) and the mixed-oxide fuel fabrication facilities (MOX-FFPs) in the fuel reprocessing cycle. Interim waste storage facilities other than AFRs also require decommissioning, but this Statement does not consider their decommissioning in detail. Estimates of costs for decommissioning these other waste storage facilities are included in total waste management costs but other effects are too small to make a significant contribution to total impacts.

Immediate dismantlement is the example decommissioning method selected here for the AFR. All of the wastes are expected to be non-TRU waste.

For decommissioning an FRP, we assume a 30-yr period of passive safe storage before dismantlement as the example method. Both TRU and non-TRU wastes are expected to result, but only the TRU portion is considered for disposal here. Most of the combustible and wet wastes generated during the safe storage period are treated with the installed waste treatment equipment, and the packaged wastes are stored in the facility until it is dismantled. The wastes generated near the end of the safe storage period, after the waste treatment facilities have been shut down, are packaged and shipped offsite to a treatment facility before being sent to disposal or storage, as are those wastes generated during the 30-yr continuing care period. The noncombustible wastes generated during dismantlement are packaged without treatment and shipped to disposal or storage.

Because of the low levels of gamma radiation, immediate dismantlement is the decommissioning method assumed here for a MOX-FFP. All of the radioactive wastes resulting from these operations are assumed to be TRU wastes. All wet wastes and most combustible wastes are assumed to be treated with the existing onsite waste treatment equipment. The combustible waste generated after the onsite waste treatment facilities have been shut down is packaged and shipped offsite for treatment prior to disposal or storage. The noncombustible waste and the treated wet and combustible wastes are packaged and shipped to disposal or storage.

Alternative decommissioning methods involving hardened safe storage were also examined for the three facilities. A continuing care period of about 100 years was considered for an AFR, while periods of about 1000 years were considered for the FRP and the MOX-FFP. The 1000-year storage period was used to provide a conservative upper bound to the environmental effects from this activity. A proposed EPA waste storage criterion would limit the safe storage period to about 100 years.

More detail on the wastes resulting from the decommissioning of these facilities is contained in DOE/ET-0028 (Section 8.0 and Section 10--Appendix A). Estimated quantities and radionuclide content of the untreated wastes from the example decommissioning processes are given in Table 4.6.1. The quantities are markedly lower than those presented earlier (Table 4.2.3) for the wastes resulting from operation of these facilities. The radionuclide content is also much lower. Quantities of packaged waste resulting from treatment of the decommissioning wastes are listed in Table 4.6.2.

The radionuclide releases estimated to occur during the decommissioning steps and during the TRU-decommissioning waste treatment operations are presented in Table 4.6.3. Except for the water from the fuel storage basins at an AFR, no release of radioactive liquids is planned. The water from the storage basins at the FRP is vaporized for discharge (using an existing vaporizer), as is the water present in the decontamination solutions.

TABLE 4.6.1. Volumes and Radionuclide Content of TRU Wastes Resulting from Decommissioning of Reprocessing Cycle Facilities

Waste Category	Facility	Volume, m^3/GWe-yr	Radionuclide Content, Ci/GWe-yr[(a)] Fission Products ^{90}Sr	^{137}Cs	Total All	Actinides ^{239}Pu	^{241}Pu	^{241}Am	Total All
Noncombustible Waste	FRP	1.4	4.7×10^{-1}	7.5×10^{-1}	2.4	7.4×10^{-3}	4.0×10^{-1}	1.1×10^{-1}	6.4×10^{-1}
(Equipment and Structural Material)	MOX-FFP	1.5	-----	----	----	2.4×10^{-1}	6.0×10^{1}	1.9	6.5×10^{1}
Compactable and Combustible Waste									
Trash	FRP	0.15	4.8×10^{-4}	7.6×10^{-4}	2.4×10^{-3}	8.4×10^{-6}	4.6×10^{-4}	1.3×10^{-4}	7.2×10^{-4}
	MOX-FFP	0.06	----	----	----	6.1×10^{-3}	1.5	4.9×10^{-2}	1.7
Filters	FRP	0.25	1.2×10^{-1}	1.9×10^{-1}	6.1×10^{-1}	5.2×10^{-3}	2.8×10^{-1}	8.0×10^{-2}	4.4×10^{-1}
	MOX-FFP	0.02	-----	----	----	2.2×10^{-1}	5.6×10^{1}	1.8	6.1×10^{1}
Concentrated Liquids, Wet Wastes, and Particulate Solids	FRP	0.15	7.9×10^{-2}	1.3×10^{-1}	4.0×10^{-1}	1.4×10^{-3}	7.6×10^{-2}	2.2×10^{-2}	1.2×10^{-1}
	MOX-FFP	0.19	----	----	----	9.0×10^{-2}	2.2×10^{1}	7.1×10^{-1}	2.4×10^{1}
Total		3.7	6.7×10^{-1}	1.1	3.4	5.7×10^{-1}	1.4×10^{2}	4.7	1.5×10^{2}

(a) At the time of assumed dismantlement (30 years after shutdown for the FRP and at the time of shutdown for the MOX-FFP), based on 30 years of facility operation before decommissioning.

TABLE 4.6.2 Estimated Quantities of Packaged TRU-Decommissioning Wastes

Waste Category	Facility	Package Type[a]	Packages/GWe-yr[b]
Noncombustible Waste (Equipment and Structural Materials)	FRP	Box	0.028
		Drum (55-gal)	6.0
	MOX-FFP	Box	0.094
		Drum (55-gal)	5.4
HEPA Filters	FRP	Drum (80-gal)	2.2
	MOX-FFP	Drum (80-gal)	0.14
Other	FRP	Drum (55-gal)	1.2
	MOX-FFP	Drum (55-gal)	0.63

(a) All packages are anticipated to have surface dose rates below 200 mR/hr, and can thus be contact-handled.
(b) Based on 30 years of facility operation before decommissioning.

TABLE 4.6.3. Radionuclides Released on Example Decommissioning of Facilities

Fission Products	Radionuclide Release[(a)] at FRP, Ci: Safe Storage	Dismantlement	TRU Waste Treatment[(b)]	Radionuclide Release[(a)] at MOX-FFP, Ci: Dismantlement	TRU Waste Treatment	Radionuclide Release at AFR, Ci: To Water Bodies	To Atmosphere[(a)]
^{90}Sr	8.0×10^{-4}	2.5×10^{-4}	7.8×10^{-10}	----	----	3.6×10^{-3}	7.2×10^{-9}
^{106}Ru	1.6×10^{-4}	----	1.6×10^{-10}	----	----	8.0×10^{-6}	1.6×10^{-11}
^{129}I	6.3×10^{-11}	4.2×10^{-11}	6.3×10^{-17}	----	----	----	----
^{134}Cs	1.3×10^{-3}	5.6×10^{-9}	2.1×10^{-10}	----	----	2.1×10^{-2}	4.1×10^{-8}
^{137}Cs	2.3×10^{-3}	4.0×10^{-4}	1.2×10^{-9}	----	----	2.2×10^{-1}	4.3×10^{-7}
^{144}Ce	1.7×10^{-4}	-----	1.6×10^{-10}	----	----	1.5×10^{-5}	3.0×10^{-11}
Total All Fission Products	7.3×10^{-3}	1.3×10^{-3}	5.1×10^{-9}	----	----	2.4×10^{-1}	4.7×10^{-7}
Actinides							
^{238}Pu	3.0×10^{-5}	2.4×10^{-8}	9.3×10^{-11}	1.2×10^{-5}	4.2×10^{-11}	----	----
^{239}Pu	2.2×10^{-6}	2.2×10^{-9}	6.8×10^{-12}	8.8×10^{-7}	3.1×10^{-12}	----	----
^{240}Pu	4.4×10^{-6}	4.5×10^{-9}	1.4×10^{-11}	1.8×10^{-6}	6.3×10^{-12}	----	----
^{241}Pu	5.6×10^{-4}	1.2×10^{-7}	1.7×10^{-9}	2.2×10^{-4}	7.6×10^{-10}	----	----
^{241}Am	2.0×10^{-5}	3.4×10^{-8}	6.2×10^{-11}	7.0×10^{-6}	2.4×10^{-11}	----	----
^{242}Cm	1.5×10^{-6}	1.9×10^{-10}	4.6×10^{-12}	----	----	----	----
^{244}Cm	2.6×10^{-5}	7.2×10^{-9}	8.1×10^{-11}	----	----	----	----
Total All Actinides	6.5×10^{-4}	1.9×10^{-7}	2.0×10^{-9}	2.4×10^{-4}	8.4×10^{-10}	----	----
Activation Products							
^{55}Fe	2.3 x 10-4	----	----	----	----	6.5×10^{-3}	1.3×10^{-8}
^{60}Co	6.5×10^{-5}	----	----	----	----	9.5×10^{-3}	1.9×10^{-8}
Total All Activation Products	6.5×10^{-4}	----	----	----	----	1.7×10^{-2}	3.3×10^{-8}

(a) Released from the facility exhaust stack.
(b) Based on the radionuclide content at the time of shutdown.

REFERENCES FOR SECTION 4.6

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4.7 ENVIRONMENTAL IMPACTS OF PREDISPOSAL OPERATIONS

Impacts of predisposal operations, including construction and decommissioning of waste management facilities and transport casks, operation of waste management facilities, and transportation of spent fuel and reprocessing wastes, are described here. Impacts considered include land, water and resource use, socioeconomic impacts, and radiological effects. The sources of this information are DOE/ET-0028 and DOE/ET-0029, which may be consulted for details.

The operational impacts discussed here are based on routine operations. Accidents and their impacts are discussed in Section 4.8. Source terms for routine releases of radioactive effluents do, however, include releases from minor accidents at reference facilities.

4.7.1 Environmental Impacts Related to Predisposal Operations for the Once-Through Fuel Cycle

The predisposal operations in the example once-through fuel cycle of this Statement include: 1) initial storage of unpackaged spent fuel in water basins either at the reactors or in away-from-reactor storage facilities (AFRs), 2) transportation of spent fuel to the disposal site (and between storage sites, if necessary), and 3) packaging of the spent fuel. An additional operation, extended storage of packaged spent fuel, is also evaluated for possible use in case there is a long delay in repository availability. The impacts of constructing, operating, and decommissioning these facilities are covered in this section.

The impacts of the fuel packaging facilities are included with those of the AFRs in this section, as in DOE/ET-0029, even though the example case for this Statement assumes that the fuel packaging facilities are located at the disposal sites. Fuel packaging facilities might also be located at the extended storage facilities, if such storage is implemented. The fuel packaging facility impacts would be essentially the same at any of the three locations.

These predisposal operations assume that the spent fuel will be disposed in a mined geologic repository within the continental U.S. The use of alternative disposal concepts could alter the number and type of predisposal facilities required. The use of a concept involving disposal outside the continental U.S. (i.e., island, subseabed, or ice sheet disposal) requires the use of additional transportation facilities (i.e., ships and docking facilities) and possible additional storage facilities. Use of the space disposal, rock melting, or well injection concepts requires the use of processing plants to obtain suitable waste forms. Impacts of such processing plants would be similar to those of a fuel reprocessing plant in the reprocessing cycle case.

4.7.1.1 Resource Commitments for Once-Through Fuel Cycle Waste Management

Land use commitments for a 3000 MTHM AFR with a fuel packaging facility are about 40 ha, of which 14 ha will be cleared for construction.

Water use will be 6×10^4 m^3 during construction and 2.5×10^5 m^3 per year during operation. As long as water can be supplied from rivers such as the reference R River (Appendix F), water use should represent a small fraction ($\sim$0.001) of the average river flow, and no significant impact will result from its withdrawal. Site selection should avoid adverse effects on aquatic systems and other downstream uses of water.

Other resource commitments during construction and operation of an AFR are presented in Table 4.7.1. Resource commitments for fabrication and use of spent fuel shipping casks are presented in Table 4.7.2.

Resource commitments during decommissioning consist mainly of steel, electricity, and diesel fuel. Total commitments of these resources during decommissioning will be small fractions of construction commitments.

TABLE 4.7.1. Resource Commitments for Construction and Operation of an Example AFR

	Construction	Operation[a]
Materials		
Concrete, m^3	2.3×10^4	---
Steel, MT	1.1×10^4	---
Stainless Steel, MT	6.1×10^3	---
Copper, MT	2.7×10^1	---
Lumber, m^3	1.3×10^3	---
Energy		
Propane, m^3	5.7×10^2	---
Diesel Fuel, m^3	5.7×10^3	---
Gasoline, m^3	3.8×10^3	---
Electricity, kWh	2.8×10^6	7.8×10^8
Manpower, man-yr	2.5×10^3	2.4×10^3

(a) Based on operation for 30 years.

TABLE 4.7.2. Resource Commitments for Fabrication and Use of Spent Fuel Shipping Casks[a]

Resource	MT/Cask	(m^3/km) per Shipment
Stainless Steel	26	--
Lead	65	--
Depleted Uranium	5	--
Diesel Fuel	--	0.0016

(a) For an "average" cask for train transport of spent fuel, which has a spent fuel capacity of about 4 MTHM.

4.7.1.2 Nonradiological Effluents of Once-Through Fuel Cycle Waste Management

Nonradiological effluents from AFR construction include dust and pollutants from machinery operation. Burning the quantities of fossil fuels listed in Table 4.7.1 also results in air pollution emissions, but concentrations in air at the fenceline from construction and operation are not expected to degrade air quality beyond applicable limits (40 CFR 50).

The major nonradiological effluent from operation of an AFR is the release of about 5×10^8 MJ/yr of heat through the cooling tower. These thermal releases are not expected to have any significant effects, nor any measurable micrometeorological effects. Predicted nonradiological effluent air concentrations from AFR operations will be considerably below applicable Federal air quality standards or naturally occurring gaseous concentrations.

Nonradiological effluents from decommissioning will be comparable to effluents during construction of the AFR and are not expected to result in any degradation of air quality.

4.7.1.3 Radiological Effects of Once-Through Fuel Cycle Waste Management

During planned operation of an AFR, the only exposure pathway to man is via airborne effluents; there are no planned releases of radioactivity to ground or water. During decommissioning, it is assumed that the purified pool water and the contained radionuclides are released to the local water bodies, however. A summary of the 70-year total body doses to the work force and the regional population during operation and decommissioning of an example AFR is given in Table 4.7.3.

In this Statement, 100 to 800 health effects are postulated to result in the exposed population per million man-rem. Based on calculated doses to the work force, 0 to 3 health effects are expected over a 70-year period as a result of operation of one 3000 MTHM AFR.

The regional population dose estimated here is a few hundred times lower than that estimated elsewhere for similar facilities (DOE/EIS-0015, Appendix B). This difference results mainly from the extra conservatism used in the other study. Both studies indicate that the doses to the regional population expected to result from AFR operation are very small in comparison to the doses to the same people during the same time period from naturally occurring sources.

TABLE 4.7.3. Doses Resulting From Operation and Decommissioning of an AFR

	70-Year Whole-Body Dose, man-rem	
	Operation	Decommissioning
Regional Population	1.4[(a)]	9.8×10^1[(a)]
Work Force	3.6×10^3	7.0×10^1

(a) The dose to the population from naturally occurring sources during the same period is about 1×10^7 man-rem.

No significant releases of radioactive material are expected during transportation of spent fuel under normal operating circumstances. However, members of the transport work force and of the population along the shipping route will receive dose from the direct radiation from the shipments. The dose for each 4 MTHM rail shipment is estimated to be 7.8×10^{-6} man-rem/km to the regional population and 5×10^{-6} man-rem/km to the transport work force. For each 0.4 MTHM truck shipment, the doses are estimated to be 2.2×10^{-6} man-rem/km to the regional population and 5×10^{-5} man-rem/km to the transport work force. For a 1,600-km shipment distance, the dose to the population for a rail shipment is 0.012 man-rem/shipment. For comparison, the estimated dose to the same population from naturally occurring sources is 230 man-rem/day.

4.7.1.4 Ecological Effects of Once-Through Fuel Cycle Waste Management

Construction of an example AFR will remove about 10 ha from its present assumed use for agriculture and wildlife for the life of the plant. While this change in land use will reduce its utility as habitat for wildlife, no significant ecological impacts to the region are expected. Disturbance of animals from fugitive dust, noise, and human activities during construction will be confined mainly to the 405-ha AFR restricted area. Erosion from run-off may deposit silt in nearby surface waters unless drainage is controlled by proper ditching, grading, and silt catchment. After construction is completed and vegetation is reestablished or surfacing is completed in the disturbed areas, the erosion problem will be reduced or eliminated.

The maximum concentrations of airborne particulates, sulfur dioxide, and carbon monoxide will occur within the 405-ha AFR restricted area. Particulate concentrations at the site during construction and decommissioning are estimated to be within Federal ambient air standards. Levels of carbon monoxide and hydrocarbons calculated to be found are only a small fraction of the existing rural air concentrations near the reference site. Concentrations of the other materials are less than applicable standards. Consequently, no measurable detrimental effects on the terrestrial ecosystem are anticipated.

During operation of the AFR, the release of about 5×10^{8} MJ/yr of waste heat is not expected to have any ecological impact. No significant effects are expected as a result of discharging the cooling tower blowdown to the local water bodies.

Particulates and gases released to the atmosphere from combustion of fossil fuels during normal transport operation are not expected to be of ecological significance.

4.7.1.5 Socioeconomic Impacts of Once-Through Fuel Cycle Waste Management

Socioeconomic impacts associated with construction and operation of an away-from-reactor storage facility depend largely on the number of persons who move into the county in which the facility will be located. Because of this, estimates were made of the size of the local population influx and their needs for locally provided social services.

The expected socioeconomic impacts of an AFR on reference sites located in the Southeast and Midwest U.S. are judged to be insignificant. The total number of estimated new in-migrants equals only about 1% of the existing population in both the construction and operation phases. In addition, there are no very large transitions over time and the expected number of in-migrants increases steadily over the life of the project.

The effect of the project is substantially different in the reference Southwest site. The number of in-migrants estimated amounts to about 9% of the existing population during construction and about 6% during operation. This decline in population influx from construction to operations of about one-third sets the stage for a boom and bust type of effect in the Southwest site.

Translating estimated project-related in-migration into socioeconomic impacts is complex and imprecise. Estimates of the level of demand that will be placed on the community to provide social services to the new workers and their families were made by applying a set of factors (Appendix G) to the project in-migration values. The product of these factors indicates how many units of each social service would be "expected" by the in-migrants. The significance of the impacts is primarily related to the capacity of the site county to meet these expectations. The calculated level of expected social services at the three reference sites is given for the year 2000 in Table 4.7.4.

TABLE 4.7.4. Selected Social Service Demands Associated with In-Migration Related to a 3000 MTHM AFR

	Expected Demand in the Year 2000		
	Southeast Site	Midwest Site	Southwest Site
Health			
Physicians	0	1	3
Nurses	1	3	9
Dentists	0	0	1
Hospital beds	1	3	11
Nursing care beds	1	3	7
Education			
Teachers	4	7	43
Classroom space, m^2 (9-12)	480	960	5180
Sanitation, m^3/day			
Water treatment	170	320	1840
Liquid waste	110	210	1260
Safety			
Firemen	0	0	2
Policemen	1	1	7
Recreation, ha			
Neighborhood parks	0	1	3
Government			
Administrative staff	0	1	3

4.7.2 Environmental Impacts Related to Predisposal Operations for the Reprocessing Fuel Cycle

Waste treatment operations required in the reprocessing fuel cycle were discussed in Sections 4.3.2 through 4.3.5 for fuel reprocessing plants (FRPs) and mixed-oxide fuel fabrication plants (MOX-FFPs). Potential waste storage requirements were discussed in Sections 4.4.2 through 4.4.4. In this section we will summarize the environmental effects of these waste management operations. The effects will be summarized for three different reference facilities: 1) a 2000 MTHM/yr FRP, 2) a 400 MTHM/yr MOX-FFP, and 3) a retrievable waste storage facility (RWSF) that has capacity to store all the high-level and TRU wastes from FRPs and MOX-FFPs during the passage of 45,000 MTHM through the fuel cycle. An RWSF will be necessary only if reprocessing is initiated significantly before a repository is available.

The environmental effects of waste treatment, storage, and transportation are summarized here for the example concepts defined in Sections 4.3, 4.4 and 4.5 for the reprocessing fuel cycle. The environmental effects of alternative concepts were also examined in DOE/ET-0029; only in the off-gas case, where the results are significantly different from those of the example concepts, are the alternatives discussed here.

The use of other than deep geologic repositories for disposal of the high-level waste could alter the number and type of waste management facilities required. As in the once-through cycle, additional transportation facilities such as ships and docking facilities would be required for disposal by the island, subseabed, or ice sheet disposal concepts. Use of the rock melting or well injection concepts to dispose of liquid waste would eliminate the need for high-level waste solidification and solidified high-level waste storage facilities but would probably require the addition of substantial liquid high-level waste storage facilities. Use of the space disposal concept would require additional chemical processing facilities and, perhaps, the addition of substantial liquid high-level waste storage facilities.

4.7.2.1 Resource Commitments in Reprocessing Fuel Cycle Waste Management

Land use commitments for waste management facilities at the reference FRP are about 19 ha compared to 60 ha for the production facilities. At the reference MOX-FFP, the waste management facilities occupy about 0.3 ha of the 6 ha required for the production facilities. An RWSF of the reference size would require 170 ha for buildings and storage areas.

Water used during construction of waste management facilities amounts to about 1.4×10^5 m^3, 5.9×10^3 m^3 and 3.1×10^5 m^3, for the FRP, MOX-FFP, and RWSF, respectively. If these quantities of water are withdrawn over the period of construction from a river such as R River, as described in the reference environment, the impact on downstream uses will be insignificant.

Resources committed for construction and operation of the waste management facilities are summarized in Table 4.7.5. Resources for construction and use of waste shipping

TABLE 4.7.5. Resource Commitments for Construction and Operation of Reprocessing Fuel Cycle Waste Management Facilities

	Waste Mgmt. Facilities at Example FRP		Waste Mgmt. Facilities at Example MOX-FFP		Example RWSF	
	Construction	Operation[(a)]	Construction	Operation[(a)]	Construction	Operation[(a)]
Material						
Concrete, m^3	7.8×10^4		3.0×10^3		2.6×10^5	
Cement, MT		3.3×10^4		1.1×10^4		2.2×10^4
Steel, MT	1.8×10^4		6.6×10^2		5.5×10^4	1.1×10^5
Stainless Steel, MT		6.6×10^3				
Copper, MT	2.0×10^2		6.9		3.0×10^2	
Lumber, m^3	5.1×10^3		1.8×10^2		1.3×10^4	
Plywood, m^2		1.0×10^5				3.0×10^5
Energy and Utilities						
Propane, m^3	1.3×10^3	8.4×10^6	4.1×10^1	3.0×10^5	3.5×10^3	
Diesel Fuel, m^3	1.2×10^4	7.2×10^2	4.2×10^2	1.6×10^3	3.5×10^4	2.2×10^4
Gasoline, m^3	8.7×10^3		3.2×10^2		2.5×10^4	
Electricity, kWh	6.4×10^6	2.7×10^9	2.8×10^5	4.2×10^7	1.7×10^4	1.4×10^9
Water consumed, m^3	1.4×10^5	1.3×10^7	5.9×10^3	2.5×10^4	3.1×10^5	9.0×10^3
Manpower, man-yr	4.0 x 103	4.5×10^3	1.9×10^2	2.6×10^2	5.1×10^3	2.6×10^3

(a) Based on operation for 30 years.

containers are given in Table 4.7.6. These resource commitments are small in comparison with those of the FRP and MOX-FFP production facilities and in an absolute sense are not expected to have a significant impact on available supplies of these materials or energy sources. Energy and materials required for decommissioning do not add significantly to the quantities of resources required for construction.

4.7.2.2 Nonradiological Effluents of Reprocessing Fuel Cycle Waste Management

Nonradioactive pollutants released to the atmosphere during construction of the FRP and MOX-FFP waste management facilities and the RWSF result from the combustion of fuel in construction vehicles and machinery, fugitive dust from ground-clearing operations, and particulates from concrete batch operations.

Offsite concentrations of carbon monoxide, hydrocarbons, and particulates resulting from construction force traffic and construction equipment emissions are projected to be less than Federal ambient air quality standards. (Onsite concentrations of particulates at the FRP and MOX-FFP construction sites were found to exceed the air quality standards; this will occur primarily as a result of construction of FRP and MOX-FFP production facilities and is a normal situation at sites of heavy construction.) Evaluation of sulfur dioxide and nitrogen oxide emissions indicates no significant effects.

The release of about 1×10^9 MJ of waste heat per year from the example FRP waste management facilities is comparable to the release of heat from a small city or town (30,000 persons) and is not expected to produce any significant effect on the environment.

Predicted concentrations of pollutants in air from waste management operations will be a small fraction of Federal air quality standards, threshold limit value concentrations

TABLE 4.7.6. Resource Commitments for Construction and Use of Waste Shipping Containers

Shipping Container	Example Capacity	Material Used in Construction, MT/cask: Stainless Steel	Material Used in Construction, MT/cask: Lead	Diesel Fuel Used per Shipment, m^3/km
High-level waste cask	Solidified HLW from 27 MTHM	25	75	0.0020
Fuel residue cask	3 fuel residue canisters (residue from 12 MTHM)	16	49	0.0013
6-drum cask	Six 55-gal drums	4	15	
14-drum cask	Fourteen 55-gal drums	5	14	
Shielded overpack	Thirty-six 55-gal drums	7	12	
Unshielded overpack	Thirty-six 55-gal drums (or equivalent volume of boxes)	7	0	0.0010

(those to which nearly all workers may be repeatedly exposed without adverse effect), and naturally occurring gaseous concentrations. Consequently, no detrimental effects are anticipated.

Water withdrawn from the R River for waste management facility operation is not expected to have adverse effects on local water supplies.

4.7.2.3 Radiological Effects of Reprocessing Fuel Cycle Waste Management

During planned operation of the waste management facilities, the only exposure pathway to man is via airborne effluents; there are no planned releases to the ground or water. For transportation of radioactive wastes under normal circumstances, no radioactive materials will be released via any pathway. However, individuals will receive doses from the direct radiation from passing rail and truck shipments.

A summary of the 70-year whole-body doses to the regional population for the individual waste management activities at the example facilities is given in Table 4.7.7.

Ninety percent of the 70-year whole-body dose to the regional population from waste management operations results from releases from the off-gas system at the FRP. The example system, which partially collects volatilized ruthenium, iodine, carbon and krypton, results in a 70-year whole-body dose to the regional population of 8300 man-rem. Should carbon and krypton be totally released, the dose would be increased to 9900 man-rem, while no treatment, i.e., release of volatilized ruthenium, iodine, carbon and krypton would increase the whole-body dose to 1.6×10^4 man-rem and result in a thyroid dose of 1×10^6 man-rem. The annual thyroid dose to the maximum individual from FRP off-gas effluents without treatment would be 0.16 rem compared to 0.002 rem with treatment. Use of the example system provides reasonable assurance that ^{85}Kr and ^{129}I releases per gigawatt-year will be within limits specified in 40 CFR 190.

The example krypton collection and storage system reduces the worldwide 70-year total body dose due to ^{85}Kr from 2.4×10^5 man-rem to 3.6×10^4 man-rem per FRP. Thus 2.0×10^5 man-rem of exposure is saved by concentrating and storing krypton. The present worth dollar cost of this savings is estimated to be $230 million; the cost per man-rem saved is thus approximately $1200. If krypton were totally released during reprocessing, the number of health effects expected to result from the ^{85}Kr radiation would be 24 to 190 per FRP. Implementation of the example krypton collection and storage system would reduce the expected number of health effects to 4 to 29 per FRP. This reduction of from 20 to 160 health effects may be compared to an estimated 60 disabling injuries and about 1 death per FRP resulting from construction of the krypton collection and storage facilities.

The 70-year whole-body dose to the worldwide population for the example treatment processes at one FRP and one MOX-FFP is 2×10^5 man-rem, which is less than 10^{-5} of the dose due to naturally occurring sources during the same 70-year period.

No significant releases of radioactive material are expected during transportation of the packaged wastes under normal operating circumstances. However, members of the transport work force and of the population along the shipping route will receive dose from the direct radiation from the shipments. These doses to the regional population are estimated to be

TABLE 4.7.7. Dose to Regional Population Due to Operation of an FRP and a MOX-FFP

	70-Year Whole-Body Dose, man-rem[a]
High-Level Wastes	
Treatment--vitrification and encapsulation	8.6×10^{2}
Storage--water basin	1.2×10^{-2}
TRU Wastes	
Treatment	
Fuel residue--package without compaction	3.5×10^{-5}
Failed equipment and noncombustible waste--package after decontamimation and disassembly of failed equipment as required.	
FRP	6.5×10^{-3}
MOX-FFP	1.2×10^{-3}
Combustible and compactable waste--incineration	
FRP contact-handled	3.3×10^{-10}
FRP remotely handled	2.8
MOX-FFP	1.6×10^{-8}
Wet wastes and particulate solids--cementation	
FRP	1.1×10^{-2}
MOX-FFP	1.7×10^{-4}
Storage	
Fuel residue--dry well	0
Other remotely handled--vault	0
Contact-handled--outdoor surface	0
Gaseous and Airborne Wastes	
Treatment	
FRP--filter and remove Ru, I, C, and Kr	8.3×10^{3}
MOX-FFP--filter	2.4×10^{-5}
Storage	
Krypton at FRP site[b]	4.0×10^{1}
TOTAL	9.2×10^{3}

(a) The whole-body dose received by the same population over the 70-year commitment period due to naturally occurring sources is 1×10^{7} man-rem.
(b) The dose due to operation of the krypton storage facility is an 80-year commitment which includes 30 years of collection plus 50 years of retention before release.

3.7×10^{-6} man-rem/km per shipment of solidified HLW or fuel residue and 1.1×10^{-6} man-rem/km per shipment of other TRU wastes. The doses to the transport work force are estimated to be 5×10^{-6} man-rem/km per shipment of solidified HLW or fuel residue and 5×10^{-5} man-rem/km per shipment of other TRU wastes. Shipments of HLW and fuel residue are assumed to be by rail and those of the other TRU wastes are assumed to be by truck.

Table 4.7.8 presents additional 70-year whole-body dose data. Included here are estimates of the doses to the work force as well as to the regional population and also the doses during transportation of the high-level and TRU wastes generated during the lifetimes of the facilities.

Doses to the work force and the regional population during decommissioning will be 10% of the 70-year total body dose resulting from operation of the facilities, assuming a safe storage period of 30 years before dismantlement of the FRP.

TABLE 4.7.8. Example Reprocessing Cycle Waste Management Operations at Individual Facilities[a]

	70-Year Whole-Body Dose (man-rem) to:	
	Work Force	Regional Population[b]
FRP Waste Management Facilities	14,000	9,200
MOX-FFP Waste Management Facilities	2,700	0.0014
RWSF	3,600	0.001
Waste Transportation	7,200	140
	27,500	9,300

(a) 30-year operation in each case.
(b) The dose to the regional population from naturally occurring sources is about 1 x 10^7 man-rem.

In this Statement, 100 to 800 health effects are postulated to occur in the exposed population per million man-rem (see Appendix E). On that basis, the 70-year total body doses to the regional population and the work force listed in Table 4.7.8, suggest that the number of health effects expected to occur as a result of waste management operations at one FRP and one MOX-FFP (plus transportation of wastes to the disposal facility) would be 2 to 20 health effects to the work force and 1 to 8 health effects to the regional population. On this same basis, the regional population dose of 10 million man-rem received from naturally occurring sources over the same 70 years suggests that 1,000 to 8,000 health effects would occur from these naturally occurring sources.

4.7.2.4 Ecological Effects of Reprocessing Fuel Cycle Waste Management

Construction of waste management facilities will remove, for the life of the plants, about 19 ha from its present use for agriculture and wildlife at the reference FRP site, and about 0.3 ha at the reference MOX-FFP site. While this change in land use will eliminate its utility as habitat for wildlife, no significant ecological impacts to the regions as a whole are expected. Disturbance of animals from fugitive dust, noise, and human activities during construction will be confined mainly to the restricted areas (2400 ha for the FRP and 400 ha for the MOX). Erosion caused by run-off may deposit silt in nearby surface waters unless drainage is controlled by proper ditching, grading, and silt catchment. After construction is completed and vegetation is reestablished or surfacing is completed in the disturbed areas, this erosion problem will be reduced.

Calculated carbon monoxide and hydrocarbon levels caused by construction of the waste management facilities are only a small fraction of the existing rural air concentrations near the reference sites. Particulate concentrations are estimated to exceed Federal ambient air standards only on the construction site. Concentrations of the other materials are below acceptable standards. Consequently, no measurable detrimental effects on the offsite terrestrial ecosystem are anticipated.

The release of heat during operation of the waste management facilities is expected to have no ecological impact. No perceptible impacts to the river ecosystem are foreseen from discharges of cooling tower blowdown. With proper intake structure design and placement in the river, the loss of aquatic organisms through intake screen impingement and entrainment in the cooling water is expected to have no significant impact on the river ecosystem.

Since the concentration of air pollutants resulting from operation of the waste management facilities is several orders of magnitude lower than those allowed by the air quality standards, no impacts to the terrestrial ecosystem are expected. No toxic effects to native plant species in the environment are expected during the life of the facilities or during decommissioning.

Some particulates and gases will be released to the atmosphere from combustion of fossil fuels during normal transport operations; however, these releases are expected to be of no ecological significance.

4.7.2.5 Socioeconomic Impacts of Reprocessing Fuel Cycle Waste Management

Socioeconomic impacts associated with waste management facilities depend largely on the numbers of persons who move into the county in which the facilities will be located. To analyze socioeconomic impacts, therefore, the size of the population influx and the needs for local social services were estimated.

The number of in-migrants resulting from construction and operation of waste management facilities is estimated to be large enough to have a significant socioeconomic impact only in the reference Southwest location for the FRP waste management facilities and the RWSF. In these two cases, the number of in-migrants amounts to about 8% of the existing population during construction and about 4% during operation. These facilities at the reference Southeast and Midwest sites are estimated to give population increases of 1% or less. The MOX-FFP waste management facilities are estimated to give population increases of 0.1% or less at each of the three reference sites.

The translation of estimated project-related in-migration into socioeconomic impacts is complex and imprecise. Estimates of the level of demand that will be placed on the community to provide social services to the new workers and their families were made by applying a set of factors (Appendix G) to the project in-migration values. The product of these factors indicates how many units of each social service would be "expected" by the in-migrants. The severity or significance of these impacts is primarily related to the capacity of the site county to meet these expectations. The calculated level of expected social services at the three sites in different areas of the U.S. is given for the year 2000 in Table 4.7.9.

The most significant demands arise for the Southwest site where an adequate labor pool is not expected to exist. However, the social service demands are small compared to those for the FRP and MOX-FFP production facilities.

TABLE 4.7.9. Selected Social Service Demands Associated with In-Migration Related to Waste Management Facilities at an FRP, a MOX-FFP, and an RWSF

	Expected Demand in the Year 2000								
	Southwest Site			Midwest Site			Southwest Site		
	FRP	MOX-FFP	RWSF	FRP	MOX-FFP	RWSF	FRP	MOX-FFP	RWSF
Personnel									
Physicians, Nurses, Dentists	1	0	1	4	0	2	10	0	8
Teachers	3	0	2	6	0	4	37	1	28
Firemen, Policemen	1	0	0	1	0	1	8	0	6
Gov't Admin.	0	0	0	1	0	0	3	0	2
Services									
Water Treatment, m^3/day	150	7	100	290	17	180	1620	23	1250
Liquid Waste, m^3/day	100	4	70	190	11	120	1080	15	840
Facilities									
Hospital and Nursing Beds	2	0	1	6	0	4	16	0	12
Classroom space, m^2 (9-12)	420	20	270	880	50	530	4480	70	3390
Neighborhood Parks, ha	0	0	0	0	0	0	1	0	2

REFERENCES FOR SECTION 4.7

U.S. Department of Energy. 1979. Technology for Commercial Radioactive Waste Management, DOE/ET-0028, Washington, D.C.

U.S. Department of Energy. 1979. Environmental Aspects of Commercial Radioactive Waste Management, DOE/ET-0029, Washington, D.C.

U.S. Department of Energy. 1980. Final Environmental Impact Statement, U.S. Spent Fuel Policy, DOE/EIS-0015, Washington, D.C.

4.8 ACCIDENT IMPACTS FOR PREDISPOSAL OPERATIONS

The environmental impacts of accidents that occur during operation of predisposal systems for both the once-through cycle and for the reprocessing cycle are described in this section. Potential accidents for the predisposal functions of treatment and/or packaging, transport, and storage are discussed here for both cycles.

The environmental impacts of accidents described in this section are representative of impacts from all postulated predisposal accidents. Using a methodology of accident identification and classification that included an umbrella source term, we selected the largest source term in classified release categories for environmental impact analysis. Results of this analysis are summarized here. Umbrella source terms are a conservative representation of releases that result from other accidents in their release category. A description of the methodology used to develop and select umbrella source terms for impact analysis is given in Section 3.2.7. Unless specified otherwise, the maximum-exposed individual in the following discussion is considered to be a member of the general public, not a radiation worker. Accident impacts are generally greater to the public than to the workers.

4.8.1 Accident Impacts for the Once-Through Cycle

This section describes the impacts of postulated accidents for handling spent fuel until it is placed in the disposal facility. Operational and long-term accident impacts from spent fuel disposal are discussed in Sections 5.5 and 5.6.

While extended storage of packaged spent fuel is not included in the example case, it may be desired if the operation of the disposal facility is delayed longer than is now expected. Therefore, analysis of accident impacts of packaged spent fuel storage are included as a contingency.

4.8.1.1 Radiological Impacts from Spent Fuel Transportation Accidents

Safety during transport of radioactive material depends primarily on shipping containers. Shipping containers must meet standards established by the Department of Transportation and the Nuclear Regulatory Commission. Containers holding significant amounts of radioactive material must prevent loss or dispersal of radioactive contents, retain shielding efficiency, ensure nuclear criticality safety, and provide adequate heat dissipation under normal conditions of transport and under specified (hypothetical) accident damage test conditions (49 CFR 173.398). Improbable accidents that exceed the severity of hypothetical tests, accidents caused by equipment failures and accidents that are less severe than the test conditions were considered in this analysis to demonstrate the range of potential occurrences in a transportation environment. Impacts of these accidents are summarized below.

Recent regulations for the shipment of spent fuel require that all shipments of spent fuel be escorted in transit; while severe accidents involving this material are still possible, the chances of occurrence will be reduced with this required increased surveillance. Chances of a period of no action by emergency response personnel following an accident, which is postulated to result in large releases of radioactive material, may be substantially reduced with these additional transportation personnel. Thus, if a severe accident does occur, consequences may be partially mitigated compared to the severe accidents described here.

Truck and rail transport of spent fuel are both expected to be used in the once-through fuel cycle. Descriptions of the systems considered in the analysis along with detailed accident descriptions are reported in DOE/ET-0028. Dose calculations for postulated accidents are reported in DOE/ET-0029. Accident frequency estimates cited in this section are based on an assumed 250 GWe nuclear industry.

The impacts examined in DOE/ET-0028 and DOE/ET-0029 were developed assuming unpackaged short-cooled (6 months out of the reactor) spent fuel. These impacts are thus much more severe than those from accidents involving long-cooled fuel. They also do not take into account the mitigation of impact that is likely to result from the new escorting regulations.

Similar accidents are also plausible for packaged spent fuel if transportation is required following packaging. However, since packaging provides an additional barrier to release of nuclides in transportation of spent fuel, the releases would be smaller and more infrequent than for unpackaged spent fuel. For this reason, specific accidents for packaged spent fuel transport are not discussed but can be assumed to cause lesser impact than unpackaged spent fuel transport.

Six accidents for truck transport of spent fuel were analyzed: three minor, two moderate, and one severe. The minor accidents involved rollovers, collisions and the undetected leakage of coolant. Only coolant leakage was expected to release radioactive material and could result in a 70-yr accumulated dose to the maximum-exposed individual of 3×10^{-6} rem at an expected frequency of approximately twice per year.

The moderate accident giving the largest release of radioactive material is a fire that activated a pressure relief valve on the cask. A 70-yr accumulated dose of 8×10^{-5} rem to the maximum-exposed individual would occur at an estimated frequency of about once every 50 years.

The severe accident culminating in a long-lasting fire results in a 70-yr accumulated dose to the maximum-exposed individual of 10 rem. The estimated frequency for this accident is about once every 50,000 years.

Eight accidents for rail transport of spent fuel were analyzed: three minor, three moderate and two severe. Two minor accidents involved derailments and 30-minute fires; no release occurred. The third minor accident involved undetected leakage of cask coolant. This accident could occur up to twice per year and result in a 70-yr accumulated dose of 2×10^{-5} rem to the maximum-exposed individual.

The moderate accidents involved cask impacts, fire-induced cask venting, and failures in the mechanical cooling system as a result of accident forces. The cooling system failure is estimated to occur once every 50 years and results in a 70-yr accumulated dose of 8×10^{-5} rem to the maximum-exposed individual.

Severe accidents resulting from extreme impacts and a prolonged loss of cooling to a design load of fuel assemblies could release significant amounts of radioactive material. Such an accident was estimated to occur once every 50,000 years. Seventy-year accumulated doses to the maximum-exposed individual of 130 rem and 140 man-rem to local populations excluding the maximum-exposed individual would result from such an accident involving 6-month cooled fuel. However, with fuel that has been cooled for several years before shipment (as planned for the once-through fuel cycle), an accident of this severity is not plausible. In a separate study of fuel transportation accidents (DOE/EIS-0015), it is reported that a maximum-exposed individual would receive a 50-yr accumulated dose of only about 0.4 rem from such an accident involving 4-yr cooled fuel (0.6 rem for a 70-yr dose).

4.8.1.2 Radiological Impacts from Unpackaged Spent Fuel Storage Accidents

The example concept for interim spent fuel storage is a 3000-MTHM capacity away-from-reactor storage facility (AFR). Eighteen accidents were postulated for the receipt and storage of unpackaged spent fuel at an AFR: eight minor, seven moderate and three severe. Accident details are described in DOE/ET-0028, Section 5.7. Eight accidents were determined to have potential for release of radioactive material. Four of the eighteen accidents relate to the operation of off-gas systems at the AFR. These accidents are not discussed here because releases from this system would be smaller than accidental releases from the dissolver off-gas system in the fuel reprocessing plant (Section 4.8.2.1) that were designated as the umbrella source terms. (Those releases result in an estimated 70-yr accumulated dose to the maximum-exposed individual of 2×10^{-3} rem.)

Releases resulting from minor accidents were added to expected annual operational releases for this facility based on their estimated frequencies.

Moderate accidents include fuel-handling mistakes, dropped transport casks and uncontrolled venting of rail casks. Releases from these accidents are smaller than those from a packaging facility accident, which is designated as the umbrella source term discussed in Section 4.8.1.3. (Those releases result in less than 3×10^{-5} rem accumulated dose to the maximum-exposed individual during the 70 years after the accident.)

A strike by a design-basis tornado, a criticality event in storage, and a loss of cooling were considered severe accidents at an AFR. The postulated criticality is estimated to occur only once every 100,000 years and results in an estimated 70-yr dose to the maximum-exposed individual of 5×10^{-2} rem.

4.8.1.3 Radiological Impacts Due to Accidents at a Fuel Packaging Facility

A fuel packaging facility (FPF) will be required to prepare fuel for disposal in the once-through cycle. The fuel packaging facility may be colocated with either the AFR, a packaged fuel storage facility or a spent fuel disposal facility. Radiological impacts that result from accidents at the packaging facility are not dependent on its location.

Six accidents were postulated for spent fuel packaging operations: three minor, two moderate and one severe. The three minor accidents involve minor fuel-handling equipment failures and are expected to result in no releases of radioactive material.

A dropped fuel element occurring about once per year was considered a moderate accident. The 70-yr dose to the maximum-exposed individual from this accident was estimated to be less than 1×10^{-5} rem.

A worst-case fuel drop accident, in which the cladding on 20% of the fuel rods is ruptured, was estimated to occur once every 100 years. This severe accident is estimated to result in less than 3×10^{-5} rem accumulated dose to the maximum-exposed individual during the 70 years after the accident.

4.8.1.4 Radiological Impacts from Packaged Spent Fuel Storage Accidents

If spent fuel is to be stored for extended periods before disposal, it may be desirable to store it as packaged spent fuel. Accidents at such facilities are discussed here. Accidents for the handling of spent fuel at a waste repository are discussed in Section 5.5.

Representative accidents for packaged spent-fuel receiving operations were considered to be similar to those postulated for a spent-fuel packaging facility (Section 4.8.1.3).

Four technologies were considered for the extended storage of packaged spent fuel: one wet and three dry. A water basin concept was considered for wet storage. Dry storage was considered in vaults, caissons and surface casks.

Nine accidents were postulated for the water basin storage of packaged fuel. Six are the result of the loss of essential basin services and would cause no release. A strike by a design-basis tornado or a criticality in the pool were considered to be severe accidents, but are expected to release less radioactivity to the environment than the equivalent accidents in the pool storage of unpackaged fuel discussed in Section 4.8.1.2 (a 70-yr dose to the maximum-exposed individual of 5×10^{-2} rem).

Various sets of severe environmental conditions were postulated for the dry storage concepts. No design-basis environments were considered capable of causing a release of radioactive material. Package failures resulting from unidentified defects or corrosion were the only mechanisms identified for material releases from dry storage. Releases are estimated to occur once every 10 years from the example facility and result in a 70-yr accumulated dose to the maximum-exposed individual of 1.1×10^{-6} rem.

4.8.1.5 Non-Radiological Impacts of Accidents in the Once-Through Cycle

Disabling injuries and deaths will result from construction of waste management facilities, as they do in construction of all facilities. Using estimates of man-hours involved in facility construction and statistical injury and death rates for censtrucion activities (13.6 disabling injuries and 0.17 deaths per million man-hours), we estimate that 110 disabling injuries and less than two deaths will result from construction of a 3000 MTHM AFR with a colocated spent fuel packaging facility. About 60% of these injuries and deaths are attributable to the AFR itself, and 40% are attributable to the packaging facility. Decommissioning activities are estimated to result in only about 3% as many deaths and injuries as do the construction activities.

Injuries and deaths will also result from spent fuel transportation, as they do from other transportation activities. For rail transport, we use estimates of 0.36 disabling injuries and 0.039 deaths per million km. For truck transport, the estimates are 0.44 disabling injuries and 0.045 deaths per million km. These injuries and deaths may occur either to the transportation worker or to the public.

4.8.2 Accident Impacts for the Reprocessing Fuel Cycle

This section describes the impacts of postulated accidents in the predisposal waste management operations required in the reprocessing fuel cycle.

4.8.2.1 Radiological Impacts from Accidents During the Treatment and Packaging of Reprocessing Wastes

In the reprocessing fuel cycle, both high-level and TRU wastes are generated at the fuel reprocessing plants (FRP), but only TRU wastes are generated at the fuel fabrication plants (MOX-FFP). Discussions of waste management accidents at these facilities are divided into high-level, transuranic, and gaseous or airborne waste management operations.

Calcination and vitrification processes were considered for the treatment of high-level liquid wastes. Minor and moderate accidents involving in-cell material spills, process equipment failures and the loss of components in the off-gas treatment processes were considered. No credible scenarios for severe accidents were identified for either of these technologies. Accidental releases are, in part, mitigated by processing through the FRP atmospheric protection system (a final exhaust-air filtration system).

The largest release from a minor accident results from a 2-kg calcine spill to the cell. Spills of this magnitude are estimated to occur once in 10 to 1000 years, but smaller spills to the cell probably will occur more frequently. The 70-yr accumulated dose to a maximum-exposed individual from this accident is 6×10^{-6} rem.

A moderate accident involving the loss of an off-gas filter is estimated to occur once every 5 years. The 70-yr accumulated dose to a maximum-exposed individual would be 2×10^{-4} rem for this accident. All other moderate accidents for the high-level waste treatment facilities would result in smaller doses.

Transuranic wastes generated in the example FRP consist of fuel hulls and hardware, failed equipment, combustible and noncombustible wastes and wet wastes. Similar wastes, with the exception of hulls and hardware, are also produced at the MOX-FFP.

Packaging without compaction, hulls compaction and hulls melting were considered for the treatment of fuel hulls and hardware. No credible moderate or severe accidents were identified for any of these technologies. The worst minor accident postulated was a zirconium fire. In this accident, 2 kg of irradiated zirconium are available for combustion. The 70-yr accumulated dose to a maximum-exposed individual was estimated to be 1×10^{-9} rem.

Failed equipment will be disassembled at both the MOX-FFP and the FRP. It is anticipated that during this operation equipment could tip over or be dropped by an overhead crane. The primary hazard from these accidents is to plant workers. No offsite releases will occur.

Combustible waste treatment technologies involve either packaging with no treatment, or controlled air incineration followed by ash immobilization. Generally, the minimum treatment processes did not have potential for other than minor accidents. Both minor and moderate accidents were identified for controlled air incinerators. No credible severe accidents were identified for the treatment of combustible wastes.

Minor accidents involving combustible wastes include minor ruptures in waste bags, small fires and waste package spills. The consequences of the largest release from a minor accident are a 70-yr accumulated dose to a maximum-exposed individual of 2×10^{-4} rem.

Moderate accidents in the incineration operation include explosions and large fires. The largest 70-yr accumulated dose from a moderate accident is 8×10^{-5} rem to the maximum-exposed individual.

Eight accidents were identified for the immobilization of wet wastes using the bitumen process: six minor and two moderate. Similar accidents are also plausible for the cementation process.

Minor accidents that do not generate areosols were considered to have no release of material beyond the processing cell area. Spillage of the treated waste product would be contained in the cell. A bitumen fire will result in the largest minor accident release. The impact of releases from this accident would be negligible.

The accident with the largest release, classified as a moderate accident, was a filter failure concurrent with a bitumen fire. This accident is expected to occur about once every 300 years and result in a 70-yr accumulated dose to the maximum-exposed individual of 5×10^{-7} rem.

There are two types of radioactive components in gaseous effluent streams. The first is radioactive gases and volatilized radionuclides. These components are captured either by adsorption beds or by cryogenic processing of the gas stream. The second is radioactive particulates entrained in the gas flow. These particulates are captured by the use of highly efficient filtration systems. Gas effluent air processing systems at the FRP may use all of these processes. However, at the MOX-FFP, filtration is the only process employed since particulates are the only significant materials in the off-gas effluent.

Minor and moderate accidents were identified for the treatment of gaseous waste streams. No credible severe accidents could be identified. Minor accidents include plugged beds and filters, minor leakage through processing equipment and failure of active system components such as blowers, pumps, etc. These accidents are considered to have no releases sufficient for consideration as an accidental release. Minor leakage was added to normal operating releases.

Moderate accidents include catastrophic filter ruptures, rupture of catalytic units during changeout and shutdown of all treatment systems. The largest release of this type would result from a shutdown of the dissolver off-gas system at the FRP for 30 days. Iodine, ruthenium, carbon and krypton would be released. A maximum-exposed individual is estimated to receive a 70-yr accumulated dose of 3×10^{-2} rem from this accident. The accident is estimated to occur about once every 10 years.

4.8.2.2 Radiological Impacts from Reprocessing Waste Storage Accidents

If waste disposal facilities are not available at the time wastes are being generated, interim storage will be required. Several storage alternatives have been analyzed for high-level waste, TRU waste, and krypton.

At the example FRP, high-level waste is solidified immediately after generation. Canisters of solidified high-level waste are then stored in water basins until they have aged sufficiently for disposal (5 years assumed). If a disposal facility is not available at that time, the waste is assumed to be sent to a sealed-cask interim surface storage facility.

Fifteen accidents were identified for water basin storage of solid high-level waste: six minor, five moderate and four severe.

Minor accidents include failure of components in ventilation and cooling systems. No releases result from these accidents.

Moderate accidents include failures of basin structural components, canister handling errors and canister failure during storage. No releases to the environment result from these accidents. Increased worker exposures are expected for accidents that release activity to the pool water.

Severe accidents in this facility involve dropping large objects into the pool, fires and a design-basis tornado strike. Consequences of these accidents are less than those cited in Section 4.8.1.2 for a spent-fuel storage pool (a 70-yr dose to the maximum-exposed individual of 5×10^{-2} rem).

The only accident with a potential for environmental consequences during sealed-cask storage of solidified high-level waste is a canister rupture during its placement in a storage cask. The accident is considered of moderate severity and, using calcine, would result in a 70-yr accumulated dose to the maximum-exposed individual of 8×10^{-3} rem.

Transuranic wastes include drums and boxes of contact-handled TRU wastes and drums and canisters of remotely handled TRU wastes, including packaged fuel residues. No credible severe accident scenarios were identified for TRU waste storage. Accidents for the storage of fuel residue are all less severe than accidents described for the cask storage of solidified high-level waste. Outdoor storage methods for all TRU wastes and indoor storage methods for remotely handled TRU wastes have potential for both minor and moderate accidents. Indoor storage methods for contact-handled TRU wastes limit the accident spectrum to minor accidents.

Typical minor accidents involving TRU waste packages include dislodging of surface contamination, rusting through of containers, and mechanical breaching of package. The 70-yr accumulated dose for the maximum-exposed individual for the largest of these releases is 2×10^{-4} rem.

Moderate accidents include fires in storage, tornado strikes and drums dropped from a crane. The 70-yr accumulated dose to the maximum-exposed individual for the largest of these releases is 4×10^{-4} rem.

Krypton removed from the FRP dissolver off gas is assumed to be collected in pressurized gas cylinders and stored onsite at the FRP in a separate facility. Three moderate accidents were postulated for the release of gas from one cylinder (130 kCi). If this occurs in the operating area or storage corridor, gas would be released via the facility stack. The 70-yr accumulated dose to a maximum-exposed individual in the public would be 5×10^{-3} rem. This accident is estimated to occur once every 20 years. Of greater potential consequence are the employee doses from this accident. A worker in the area of the ruptured cylinder faces hazards from flying debris and could receive a radiation dose rate of up to 8 rem/min. Immediate evacuation of the area would be required.

4.8.2.3 Radiological Impacts from Reprocessing Waste Transportation Accidents

A reprocessing fuel cycle has potential transportation requirements for spent fuel, solidified high-level waste, fuel residues, and other TRU wastes. As in the once-through cycle, safety during transport depends primarily on shipping containers. The containers must meet standards established by the Department of Transportation and the Nuclear Regulatory Commission. Packages containing significant amounts of radioactive material must be designed to prevent loss or dispersal of the radioactive contents, retain shielding efficiency, ensure nuclear criticality safety, and provide adequate heat dissipation under normal conditions of transport and under specified (hypothetical) accident damage test conditions (49 CFR 71, Appendix B). Improbable accidents that exceed the hypothetical tests, accidents due to equipment failures and accidents that are less severe than the test conditions were considered here to demonstrate the range of potential occurrences in a transportation environment.

Minor, moderate and severe accidents were postulated for the rail transport of solidified high-level waste. Minor accidents for this material are similar to those for spent fuel. A moderate accident could result in a reduction in neutron shielding and a local hazard of increased neutron exposures. No radioactive material would be released in this accident. A severe accident involving impact and fire could result in a material release. This accident is estimated to occur only once every 330,000 years and result in a 70-yr accumulated dose to the maximum-exposed individual of 10 rem.

Transuranic wastes were considered to be transported in DOT-licensed packages. Three minor and one severe accident were identified. The worst minor accident is expected to occur once per year due to improperly closed waste packages and result in a 70-yr accumulated dose to the maximum-exposed individual of 3 x 10^{-3} rem. A severe accident involving severe impact and fire with an estimated frequency of once every 100,000 years would result in a maximum-exposed individual 70-yr whole body dose of 3 rem.

4.8.2.4 Non-Radiological Impacts of Accidents in the Reprocessing Cycle

Estimates of deaths and disabling injuries resulting from construction and decommissioning of reprocessing fuel cycle waste management facilities are given in Table 4.8.1. Injuries and deaths also result from transportation of the wastes. As in spent fuel transport, we use estimates of 0.36 disabling injuries and 0.039 deaths per million km for rail transport and 0.44 disabling injuries and 0.045 deaths per million km for truck transport. These injuries and deaths may occur either to the transportation worker or to the public.

TABLE 4.8.1. Disabling Injuries and Deaths from Construction and Decommissioning of Reprocessing Fuel Cycle Waste Management Facilities

Construction	Disabling Injuries[(a)]	Deaths[(b)]
Waste Mgmt. Facilities at Example FRP	55	0.7
Waste Mgmt. Facilities at Example MOX-FFP	5	0.06
Example RWSF	415	5
Decommissioning		
Waste Mgmt. Facilities at Example FRP	25	0.3
Waste Mgmt. Facilities at Example MOX-FFP	5	0.06

(a) Based on frequency rate of 13.6 per million man-hours.
(b) Based on frequency rate of 0.17 per million man-hours.

4.8.3 Radiological Impact Summary for Predisposal Operations Accidents

Table 4.8.2 summarizes the radiation effects of the predisposal-system accident analysed for this Statement.

This comparison shows that transportation is the waste management step with the potential for the most serious accident in either fuel cycle. The estimated exposures in these accidents, however, are not large enough to cause observable clinical effects. The individuals exposed would presumably bear an increased probability of developing cancer sometime during their life or of passing on a genetic defect.

TABLE 4.8.2. Summary of Radiation Effects from Potential Worst-Case Predisposal System Accidents

	Maximum-Exposed Individual Radiation Doses, rem	
	Once-Through Cycle	Reprocessing Cycle
Transportation		
Spent Fuel (4-yr-old)	0.6[a]	
HLW		10[b]
TRU Waste		3
Storage	5×10^{-2}	8×10^{-3}
Treatment and Packaging	3×10^{-5}	2×10^{-3}

(a) Shipment of 6-month-old spent fuel, which is unlikely, could result in a maximum dose of 130 rem.
(b) Based on HLW 6.5 years after reactor discharge.

REFERENCES FOR SECTION 4.8

Code of Federal Regulations, 49 CFR-173.398.

U.S. Department of Energy. 1979. Technology for Commercial Radioactive Waste Management, DOE/ET-0028, Washington, D.C.

U.S. Department of Energy. 1979. Environmental Aspects of Commercial Radioactive Waste Management, DOE/ET-0029, Washington, D.C.

U.S. Department of Energy. 1979. Final EIS on U.S. Spent Fuel Policy for the Storage of U.S. Spent Power Reactor Fuel, DOE/EIS-0015, Washington, D.C.

4.9 COST OF PREDISPOSAL OPERATIONS

Costs for treating, storing, and transporting spent fuel or commercial reprocessing and mixed oxide fuel fabrication wastes are presented in this section. All costs are stated in terms of constant[a] 1978 dollars.

The costs shown here are levelized[b] unit costs based on capital, operating, and decommissioning costs for the individual predisposal waste management operations. Capital, operating, and decommissioning cost estimates have been developed as part of this Statement for the predisposal facilities associated with the example geologic disposal system and are summarized in Appendix A. Predisposal costs for alternatives other than geologic disposal are based on predisposal costs of the geologic disposal system where the operations are similar. Where the operations are different, data from other studies have been used to the extent available.

For the once-through cycle, the mined geologic and very deep hole concepts have the lowest predisposal systems costs ($103/kg HM) of the alternatives studied in this Statement. Costs of other alternatives are 50 to 100% higher. For the reprocessing cycle, the mined geologic, very deep hole, well injection, space injection, and rock melting alternatives all cost about $170/kg (including spent fuel storage and transportation). Costs of other alternatives ranged from $15 to over $230/kg HM more than the lowest cost options.

The cost tables in this section are intended to provide predisposal cost comparisons between disposal alternatives and to illustrate cost relationships among predisposal components for the example geologic disposal alternative. The total costs presented here do not include the significant costs of research and development. Costs for the entire waste management system, levelized with respect to the power generation that produced the waste, are developed in Chapter 7.

A brief explanation of the cost estimate assumptions and bases for the costs developed in this Statement is given in Section 3.2. Additional detail on predisposal facility costs for geologic disposal is available in DOE/ET-0028, Volumes 2, 3 and 4.

4.9.1 Once-Through Fuel Cycle Predisposal Costs

For the example once-through cycle, predisposal operations consist of storage at reactor basins, storage in independent basins when reactor basin capacities are exceeded, treatment and packaging of the fuel assemblies, and all transportation operations. A brief description of the operations required for each disposal option is found in Table 4.1.1.

Table 4.9.1 lists the costs associated with these predisposal operations for the alternative disposal methods studied. Reactor basin storage charges of $25/kg HM and transportation costs of $26/kg HM for shipment of spent fuel to treatment facilities are common to all

(a) For a definition of constant dollar costs, see Section 3.2.8.1.
(b) Levelizing refers to developing a single, constant unit charge, which recovers all expenditures associated with a facility or system including interest (see Section 3.2.8.2). The costs stated in this section are levelized with respect to individual waste management operations only.

TABLE 4.9.1. Unit Costs of Predisposal Operations for Once-Through Cycle Disposal Options

Predisposal Operation	Cost, $/kg HM: Mined Geologic	Very Deep Holes	Rock Melting	Island	Sub-seabed	Ice Sheet	Injection Well	Transmutation	Space Injection
5-Year Reactor Storage	25	25	25	25	25	25	25	25	25
Shipment to Interim Storage (1000 mi)[a]	5	5	9[c]	5	5	5	9[c]	9[c]	9[c]
Interim Storage[a]	29	29	39[c]	29	29	29	39[c]	39[c]	39[c]
Shipment to Treatment (1500 mi)	26	26	26	26	26	26	26	26	26
Treatment and Packaging	18	18	70[c]	18	18	18	70[c]	200[c]	~100[c]
Shipment to Disposal	--[b]	--[b]	6	50	50	50	6	20	<15
TOTAL	103	103	175	150	150	150	175	320	<214

(a) Based on interim storage of 25% of total spent fuel discharges.
(b) No cost is shown for this step since the analysis assumes that packaging or treatment is accomplished at the disposal site. If packaging facilities for mined geologic disposal of spent fuel were located offsite, an additional transportation step would be necessary for this option.
(c) Includes costs of managing TRU wastes generated during dissolution of the spent fuel.

disposal alternatives. The rock melting, well injection, transmutation, and space injection alternatives have somewhat higher costs for shipment to interim storage, interim storage, and treatment since the spent fuel is dissolved and management of additional waste streams is required. The high transportation costs of the island, subseabed, and ice sheet alternatives are a result of the required land and ocean transportation.

The mined geologic and very deep hole concepts have significantly lower predisposal costs than the other alternatives, $103/kg HM. The island, subseabed, and ice sheet alternatives have higher costs, $150/kg HM, because of the expensive transportation requirements. The other alternatives have higher predisposal costs because of the cost of managing the additional waste streams generated. These range from $175/kg HM for the rock melting and well injection alternatives to $320/kg HM for transmutation.

4.9.2 Reprocessing Fuel Cycle Predisposal Costs

A brief description of the predisposal operations for the reprocessing fuel cycle required for each of the disposal options is found in Table 4.1.2. Costs associated with these operations are shown in Table 4.9.2. Spent fuel storage and transportation costs could be considered as reprocessing costs rather than as waste management costs if spent fuel is reprocessed. For consistency and conservatism, the costs of spent-fuel storage and shipment are included as waste management costs in this Statement. Without these costs, the predisposal costs of the reprocessing cycle alternatives are comparable to or less than the once-through cycle costs.

Waste treatment costs of the reprocessing cycle alternatives are comparable with two exceptions: 1) costs for the rock melting and well injection alternatives are lower since high-level waste solidification is not required, and 2) costs for the transmutation alternative are higher because of repeated chemical partitioning and target fabrication operations.

Transportation costs for the rock melting and well injection alternatives are less than other options since the high-level waste is not transported offsite. However, the cost of interim storage of the high-level liquid waste for these two alternatives is much higher than the cost of solidified high-level waste storage employed in the other alternatives. Transportation costs for the island, subseabed, and ice sheet alternatives are significantly higher than for other alternatives because of the oceanic shipments of high-level waste.

Total predisposal system costs of the mined geologic, very deep hole, rock melting, well injection, and space injection alternatives are similar, e.g., $168/kg HM. The costs of the island, subseabed, and ice sheet alternatives are 185/kg HM or about 10% higher and costs of the transmutation alternative (>$400/kg HM) are more than 100% higher than any other alternative.

TABLE 4.9.2. Unit Costs of Predisposal Operations for Reprocessing Waste Disposal Operations

Predisposal Operation	Mined Geologic	Very Deep Holes	Rock Melting	Island	Sub-seabed	Ice Sheet	Injection Well	Trans-mutation	Space Injection
	Cost, $/kg HM								
Spent Fuel Storage and Shipment	59	59	59	59	59	59	59	59	59
Waste Treatment									
• FRP[a,c]	67	67	43	67	67	67	43	>230[f]	~67[e]
• MOX-FFP[b]	4	4	4	4	4	4	4	>70[f]	4
Shipment to Interim Storage (1000 mi)	6	6	4	6	6	6	4	6	6
Interim Storage[d]	23	23	52	23	23	23	52	23	23[e]
Shipment to Disposal (1500 mi)	9	9	6	26	26	26	6	>12	<15
	168	168	168	185	185	185	168	>400	<174

(a) Fuels Reprocessing Plant. See Appendix A for a breakdown of example FRP waste treatment costs and options.
(b) Mixed Oxide Fuel Fabrication Plant. See Appendix A for a breakdown of example MOX-FFP waste treatment costs and options.
(c) Includes HLW and TRU waste treatment costs ($/kg HM) as follows:

	Mined Geologic and Similar Cost Options	Rock Melting	Injection Well	Transmutation	Space Injection
HLW	24	--	--		~24
TRU Waste	43	43	43		43
TOTAL	67	43	43	>>230	~67

(d) A $10/kg HM cost for TRU waste storage is common to all options. The remaining cost is for HLW storage.
(e) HLW storage costs for those options may differ from those for the mined geologic option because of different configurations. No difference is assumed here.
(f) Based on additional partitioning facility costs.

4.9.3 Detailed Predisposal Cost Estimates for Geologic Disposal

This section describes in greater detail the predisposal cost estimates for the example geologic disposal alternative. Costs are derived for both the once-through and reprocessing cases.

4.9.3.1 Once-Through Fuel Cycle

Table 4.9.3 lists the costs associated with once-through predisposal operations. Reactor basin storage is estimated to cost about \$6/kg HM per year with storage periods on the order of five years, for an equivalent present-worth cost of about \$25/kg HM.

After storage, the fuel assemblies may be: 1) packaged intact, 2) disassembled and packaged, 3) chopped, voloxidized, and packaged, or 4) chopped, the fuel dissolved, and converted to glass. Treatment costs shown in Table 4.9.3 for the above options range from \$18 to \$92/kg HM due to the increasing complexity of these operations.

Costs for independent unpackaged water basin storage of spent fuel vary significantly with the size and capacity utilization of the facility. Costs for storage in a non-expandable 3000 MTHM basin are estimated at about \$117/kg HM.[a] Costs for a 5000 MTHM non-expandable basin (DOE 1978), using unit cost assumptions in this Statement are estimated at \$80/kg HM.[a] Estimates for a facility expandable to 20,000 MTHM are \$45/kg HM.[a] In addition, costs vary nearly inversely with capacity utilization. For example, if a facility utilized only 50% of its capacity, unit costs would be almost doubled.

Other storage options include storage of packaged spent fuel. In these cases, spent fuel could be packaged in facilities located adjacent to storage facilities. Table 4.9.3 illustrates costs for four such design concepts. Dry well storage appears to be the most cost effective alternative.

Packaging of the spent fuel could be done either at facilities adjacent to storage basins or at the repository. Packaging facilities that are integral with the repository are assumed for the example system here and may be more cost effective due to lower transportation costs for unpackaged spent fuel.

Transportation costs include transport of the spent fuel from reactor storage to independent storage (25% of the fuel), reactor storage to repository (75% of the fuel) and independent storage to repository (25% of the fuel).

Total predisposal costs for the example case in Table 4.9.3 are about \$103/kg HM. The range is estimated using the lowest and highest cost options.

(a) In the cases shown in Table 4.9.3, it is assumed that only about 25% of total spent fuel discharges are sent to independent storage and the cost is reduced proportionally.

TABLE 4.9.3. Predisposal Unit Costs for the Once-Through Cycle

Treatment	Unit Cost, \$/kg HM[a] Example System	Other Options
Decay Storage at Reactor Basin	25[b]	--
Package Intact Fuel Assemblies	18	--
Disassemble and Package Fuel Rods	--	38[c]
Package Chopped and Voloxidized Fuel	--	42[c]
Dissolve Fuel and Convert To Glass	--	92[c]
Independent Away-from Reactor (AFR) Fuel Storage		
Unpackaged		
• Nonexpandable 3000 MT Basin	29[d]	--
• Nonexpandable 5000 MT Basin	--	20[d]
• Modular Basin Expanded to 20,000 MT	--	11[d]
Packaged		
• Water Basin	--	38
• Air-Cooled Vault	--	35
• Dry Well	--	22
• Surface Cask	--	30
Transportation	31[d,e]	--
Total	103	range 85 to 186

(a) Costs may be expressed in \$/GWe-yr by multiplying by 38,000 kg HM/GWe-yr.
(b) Reactor basin spent fuel storage costs are based on a charge of \$6/kg HM per year. The value shown in the table is equivalent to a minimum storage time of 5 years with a real cost of money of 7% per year.
(c) Estimates based on facilities and operations described in ONWI-39, July 1979, except that the cost calculations were modified to a 7% real cost of money basis. Estimates include treatment of all wastes generated, but do not include transportation and disposal. Costs for the entire system are shown in Table 4.9.7.
(d) Average fuel cycle cost based on interim storage of 25% of total spent fuel discharges.
(e) Packaging may be done at the repository or at another site. The transportation costs for the example case are based on a packaging facility which is integral with the repository and assumes that packaged fuel handling is accomplished using repository facilities. Transportation costs consist of \$5/kg HM for shipment of 25% of the spent fuel to AFR storage, \$20/kg HM for shipment of the other 75% of the spent fuel from reactor basins to final disposal and \$6/kg HM for shipment of the fuel in AFR storage to final disposal.

4.9.3.2 Reprocessing Fuel Cycle

Reprocessing fuel cycle wastes consist of wastes from reprocessing and mixed oxide fuel fabrication plants. Table 4.9.4 shows the unit costs for alternative methods of waste treatment for these wastes.

Differences in cost between treatment options are not large, ranging from 10 to 25%, except for krypton removal. Predisposal costs for the example system are fairly evenly distributed between high-level waste treatment (\$23.9/kg HM), TRU waste treatment (\$18.40/kg HM), gaseous waste treatment (\$28.20/kg HM), interim storage of high-level and TRU wastes (\$23.10/kg HM) and transportation (\$15.50/kg HM). These costs total about

TABLE 4.9.4. Unit Cost Estimates for Reprocessing Fuel Cycle Wastes

Waste Category	Treatment	Unit Cost, \$/kg HM[a] Example System	Other Options
High-Level Liquid Waste	Spray Calcination & Vitrification	10.4	
	Fluid Bed Calcination Only		13.0
	5-Year Onsite Storage & Handling (after solidification)	13.5	
Fuel Residue	Package Without Compaction (in sand)	4.9	
	Compaction of Hulls		4.6
	Melting of Hulls		5.1
Non-Combustibles and Failed Equipment	Package	4.8[b]	
Combustible and compactable	Incineration	4.4[b]	
	Package Only		3.7[b]
Wet Waste	Cementation	4.3[b]	
	Bitumenization		4.3[b]
Gaseous Waste	Vessel Off-Gas	3.9	
	Dissolver Off-Gas		
	I and Ru Removal		2.0
	I, Ru and C Removal		3.2
	I, Ru and Kr Removal		6.0
	I, Ru and C and Kr Removal	6.1	
	Kr Storage Onsite	16.4	
	Atmospheric Protection System (APS)		
	Group III Prefilter	1.8	
	Sand Filter		3.8
	Deep Bed Filter		2.5
Solidified Reprocessing Wastes	Interim Storage[c]	23.1	
	Transportation[d]	15.5	
	Subtotal	109	
	Spent Fuel Storage and Shipment Prior to Reprocessing	59[e]	
Total		168	range 139 to 179

(a) Costs may be expressed in \$GWe-yr by multiplying by 38,000 kg HM/GWe-yr.
(b) Includes estimates for waste treatment at the mixed oxide fuels fabrication plant. See Appendix A for further detail.
(c) See Table 4.9.5.
(d) See Table 4.9.6.
(e) Based on a 1-year storage of all spent fuel at the reactor basin (\$6/kg HM) and interim storage of 25% of total spent fuel discharges (\$29/kg HM). Spent fuel transportation is estimated to cost \$24/kg HM (see Table 4.9.6). Although spent fuel handling and storage prior to reprocessing are not clearly waste management functions, the costs are shown here and are included in the systems cost estimates in Chapter 7 to conservatively estimate waste management costs.

\$109/kg HM, which is comparable to the \$103/kg HM predisposal cost totals for spent fuel waste management. In addition, spent fuel handling and storage costs before reprocessing are also included for reasons noted previously, bringing the total reprocessing fuel cycle waste management cost to \$168/kg HM. The range is estimated using the lowest and highest cost treatment options.

Tables 4.9.5 and 4.9.6 show additional detail for the costs of interim storage options and transportation operations.

4.9.4 Detailed Subsystem Costs for Geologic Disposal

Since many treatment options affect the treated waste volumes, the entire cost impact of these options cannot be evaluated on the basis of the predisposal costs alone. For this reason final disposal costs are included in the subsystem costs presented here, although they are not developed in this Statement until Section 5.6.

Table 4.9.7 illustrates total subsystem waste management costs for waste management operations for both the once-through and reprocessing fuel cycles. These costs include the effect of volume reduction on subsequent transportation, interim storage and disposal operations. For the once-through cycle, dissolving the spent fuel costs significantly more than other treatment options. For the reprocessing cycle, treatment options do not have a significant impact on total system costs except for the fuel residue and combustible waste options. The high cost of removing and storing krypton relative to the waste management costs of removing other gases can also be noted.

The cost ranges reflect the impact of volume changes on costs assuming the example interim storage and final disposal methods. The upper cost estimate assumes the least volume reduction and the lowest cost estimate the greatest volume reduction. Cost ranges would be somewhat greater than shown here for other interim storage and final disposal options.

The example total cost estimate of \$215/kg HM for the reprocessing fuel cycle includes \$59/kg HM for spent fuel transportation and storage prior to reprocessing. The final disposal costs included in the subsystems cost estimates may be estimated by subtracting the predisposal costs in Tables 4.9.3 and 4.9.6 from the subsystem cost in Table 4.9.7 for the once-through and reprocessing fuel cycles.

TABLE 4.9.5. Unit Cost Estimates for Interim Storage Operations for Reprocessing Fuel Cycle Wastes

Waste Category	Operation	Unit Cost $/kg HM[a] Example	Option
High-Level Waste	Sealed Cask Storage	13	--
Fuels Residue and Other TRU Waste Canisters	Dry Well Storage	7	
	Vault Storage		20
Remotely Handled TRU Waste Drums	Vault Storage	3	
	Dry Well Storage		6
Contact-Handled TRU Waste Drums and Boxes	Outdoor Surface Storage	0.3	
	Indoor Unshielded Storage		0.4
Total		23	

(a) Costs may be expressed in $/GWe-yr by multiplying by 38,000 kg HM/GWe-yr.

TABLE 4.9.6. Unit Cost Estimates for Example Transportation Operations, $/kg HM[a]

Origin and Destination	Unit Cost for Spent Fuel	Unit Cost For Reprocessing Wastes: High-Level Waste	Fuel Residue Waste	Other RH-TRU Waste	CH-TRU Waste
Reactor to Interim Storage (1000 mi)	5[b]	--	--	--	--
Reactor to Reprocessing Plant (1000 mi)	14[c]	--	--	--	--
Interim Storage to Reprocessing Plant (1000 mi)	5[b]	--	--	--	--
Reprocessing Plant to Interim Storage or Repository (1000 mi)	--	2.0	2.2	1.7	0.2
Interim Storage to Repository (1500 mi)	--	3.0	3.5	2.6	0.3
Total	24	5	6	4	0.5

(a) Costs may be expressed in $/GWe-yr by multiplying by 38,000 kg HM/GWe-yr.
(b) Based on interim storage of 25% of the spent fuel.
(c) Based on direct shipment of 75% of the spent fuel.

TABLE 4.9.7 Subsystems[a] Waste Management Costs for Alternative Waste Treatment Options

Fuel Cycle Option	Waste Category	Option	Systems Cost,[a] $/kg HM[b] Example	Options
Once-Through	Spent Fuel	Encapsulate Whole Assemblies	155	
		Disassemble and Encapsulate		~140
		Chop Assemblies and Encapsulate		~150
		Dissolve Fuel, Convert to Glass and Encapsulate		~250
	Total		155[c]	range 140 to 250
Reprocessing	High-Level	Vitrification	66	
		Calcination		69
	Fuel Residue	Package in Sand Without Compaction	20	
		Compaction of Hulls		14
		Melting of Hulls		11
	Non-combustible and Failed Equipment	Package	20	
	Combustibles	Incinerate	10	
		Package Only		41
	Wet	Cementation	12	
		Bituminization		8
	Gaseous	Vessel Off-gas	4	
		Dissolver Off-gas		
		• I and Ru Removal		2
		• I, Ru and C Removal		3
		• I, Ru and Kr Removal		22
		• I, Ru, C and Kr Removal	22	
		Atmospheric Protection System		
		• Group III Prefilter	2	
		• Sand Filter		4
		• Deep-Bed Filter		3
	Subtotal		156	
	Spent Fuel	Storage and Transportation Prior to Reprocessing	59	
	Total		215[d]	range 182 to 251

(a) Subsystems costs include the cost of waste treatment, packaging, all transportation, interim storage and final disposal. Research and development costs and the discount rate effect of timing of the costs are not included in the figures shown here, but are included in the system power cost estimates in Chapter 7.
(b) Costs may be expressed in $/GWe-yr by multiplying by 38,000 kg HM/GWe-yr.
(c) Includes $52/kg HM for geologic disposal.
(d) Includes $47/kg HM for geologic disposal.

REFERENCES FOR SECTION 4.9

Office of Nuclear Waste Isolation. 1979. An Assessment of LWR Spent Fuel Disposal Options, ONWI-39, Office of Nuclear Waste Isolation, Battelle-Columbus Laboratories, Columbus, Ohio.

U.S. Department of Energy. 1978. Conceptual Design Report for the Away from Reactor Spent Fuel Storage Facility, DPE 3547. Prepared by E.I. Du Pont De Nemours and Co., Inc. Wilmington, Delaware.

4.10 SAFEGUARDS INCLUDING PHYSICAL PROTECTION FOR PREDISPOSAL OPERATIONS

Regulations similar to those already in place to protect the public from theft of nuclear material and from sabotage at licensed nuclear facilities are expected to apply to operations at waste management facilities. The probable safeguard requirements for predisposal waste management facilities are described in this section.

4.10.1 Safeguards Requirements for the Once-Through Cycle

Safeguards measures, including physical protection, required for currently licensed nuclear facilities are expected to be adequate for safeguards and physical protection for the once-through cycle. Spent fuel and the facilities designed to manage this material are not expected to require additional safeguards.

4.10.1.1 Spent Fuel Treatment and Packaging Safeguards Requirements

The susceptibility of the spent fuel handling operation to theft and sabotage of the fuel elements is reduced as packaging and treatment operations of the fuel elements proceed. The spent-fuel elements and all treatment and packaging facilities handling this material will be physically protected as required by Federal regulations for vital areas (see Section 3.2.9 or 10 CFR 70, 73). All of the auxiliary systems for spent fuel handling will be similarly protected because they are part of the same facility.

If the spent fuel is simply encapsulated for disposal as in the example process for this Statement, the spent fuel elements become less attractive and less accessible targets for sabotage. In addition, operating safety features inherent in the design of facilities licensed to process spent fuel elements contribute significantly to safeguarding this material.

If the spent fuel is chopped and encapsulated, none of the additional steps required in this process significantly increase the susceptibility of the facility, equipment or target material to theft or sabotage.

If the spent fuel is dissolved and converted to glass, the physical protection requirements and the relative unattractiveness and inaccessibility of the material make it an unlikely target for theft or sabotage. The same protective environmental and control measures identified above are present in this facility to provide required safeguards features.

4.10.1.2 Safeguards Requirements for Spent Fuel Storage

Spent fuel is neither easily accessible nor an attractive enough source of fissile material to encourage theft. The plutonium concentration is low and the fuel elements are very radioactive; massive shielding of steel, lead, concrete or several feet of water is required at all times. Separation of the plutonium requires complex chemical processes carried out in remotely operated, shielded processing equipment. In addition, spent fuel is not in a form suitable for easily dispersing radioactive material, and thus, is not an attractive target for this threat because only intact spent fuel rods are considered to be an acceptable form for extended storage.

Physical protection features required by Federal regulations are expected to provide adequate safeguards. Safeguards contingency plans in these regulations for licensed facilities will include NRC-approved arrangements for support from local law enforcement personnel if there is a serious threat. An adequate response force will be able to engage and contain the intruders in less time than is required for the intruders to gain access, remove fuel elements from the storage location, transfer them to a shielded container, place them on a vehicle and leave the site. A single fuel assembly weighs more than one-quarter metric ton and a hoist or crane operated from behind heavy shielding is required to move it. Disassembly to obtain individual fuel rods, which could be transferred by more readily obtainable light equipment, would be a much more time-consuming operation. The disassembly would have to be done remotely, behind heavy shielding or under water.

These same measures also deny fuel storage facility access to saboteurs. A detailed study (Voiland et al. 1974) of the safeguards risks associated with water basin storage of spent fuel concluded that the stored irradiated fuel at the facility under consideration is not amenable to a credible sabotage event that would endanger the public health and safety. The safeguards measures assumed for that case are typical of those required for the licensed facilities.

4.10.1.3 Safeguards Requirements for Transport of Spent Fuel

Spent fuel is more vulnerable to theft and acts of sabotage during transport than at fixed sites because it is more accessible. The measures proposed to protect against diversion and sabotage of shipments of spent fuel reflect this potential threat (10 CFR 73).

The level of physical protection required for shipments of spent fuel elements, established by the NRC in an interim rulemaking (10 CFR 73 1979), was based on a study by Sandia laboratories (1977). Specific requirements were included to protect the public against sabotage of spent fuel in transit by truck or rail, with particular concern for urban areas.

Theft of spent fuel to obtain the fissionable material is not sufficiently credible to warrant additional requirements for this specific threat (see Section 4.10.1.2). Theft of this material as a part of an extortion attempt would be limited to the length of time law enforcement personnel would need to locate the stolen cask. Such material in a cask is detectable by aerial radiation surveys and the fact that detection would be imminent would deter any lengthy extortion scheme.

Prediction of detection is based upon the capability of the Department of Energy's Aerial Radiological Monitoring system (ARMS) of which two are in continuous service (Doyle

1976). It is assumed that one of these or an equivalent system would be available. The system consists of a forty-sensor array with a computer-assisted data analyzer, a printer and a plotter mounted in a helicopter.[a]

4.10.2 Safeguards Requirements for the Reprocessing Fuel Cycle

In the reprocessing fuel cycle large quantities of fissionable and radioactive material are handled in a fuel reprocessing facility, and the physical protection requirements for the facility and vital materials within it are specified in 10 CFR 73. The general features of those requirements are identified in Section 3.7. Similar requirements would be enforced at the plutonium-uranium mixed oxide fuel fabrication plants.

The waste materials produced at these facilities are unattractive as targets of theft compared to the fissionable material in the facilities. In addition, all waste treatment operations and storage of highly radioactive wastes would be protected in "vital" areas. Consequently, these materials would be inaccessible to any but authorized persons, and successful intrusion, theft and sabotage are improbable.

4.10.2.1 Safeguards Requirements for the Treatment of Reprocessing Wastes

High-level waste is not a potential source of fissionable material and could only be a target for theft or sabotage to disperse or threaten dispersal of radioactive material. The HLW is an unattractive target because of its high radiation level and inaccessibility. All handling, storage, and treatment in the facility occurs by remote operations in shielded, isolated vessels and cells.

Before it is solidified, HLW may be stored as a solution in shielded tanks in which it is accessible only by remote means. Its intense radioactivity and high heat release rates and the maze of facility support equipment would make unauthorized transfer of HLW to a shielded container and its removal offsite an incredible accomplishment, particularly since extensive physical protection measures would also have to be overcome. For similar reasons, dispersal of HLW onsite by explosives is not credible, although the concentration of radioactive fission products in this waste may make it appear to be an attractive target.

With inside assistance, physical protection and access control measures could possibly be compromised, and sabotage of the storage facility could occur. One consequence could be a disruption of the waste cooling system and/or electrical system. Self-heating would cause the contents to begin to boil in about 7 hours and boil to dryness in about 100 hours. This scenario is not considered credible if the planned safeguards measures and the safety design features of the facility (which are included to ensure continuity of HLW cooling) are

(a) The ability of an aerial radiation survey to detect a spent fuel cask that has not been breached and is located inside a facility depends upon the facility. In a single-storied, conventionally constructed warehouse or its structural equivalent, the radiation from the cask would be readily detected in an aerial survey. If the truck and fuel cask were in an underground garage under a multi-storied building surrounded by multi-storied buildings, an aerial survey may not be effective. However, a mobile surface survey would be effective in detecting this source.

considered. Some facility damage and a 300-ℓ (80-gal) spill to the ground during a 3-hour period are considered to be representative of the most serious results from the worst act of sabotage (see DOE/ET-0028, p. 5.1.37).

Solidified HLW from the reprocessing cycle, which contains nearly all of the fission products and very little plutonium, could conceivably be a target of theft for a subsequent threat of dispersal of the radioactive material. However, the handling problems during attempted theft are as formidable for HLW as for spent fuel. Heavy shielding and special equipment are required to avoid serious radiation exposure. These factors make HLW relatively unattractive for theft for any purpose, regardless of the form.

The TRU wastes would also be processed or treated in vital areas until they have been concentrated and/or packaged so they can be transported and stored without hazard. After packaging, the low radiation items may be stored onsite in protected, access controlled areas. The materials in packaged form contain only small amounts of fissionable material, and are unattractive targets for theft. Sabotage would require access to the storage location in the plant. If sabotage is successful, the facility may be damaged and the site contaminated with radioactive waste. The contamination is expected to be contained with little or no public exposure because of the plant location, site layout, and safety features.

The principal products of the example dissolver off-gas treatment facilities are the radionuclides krypton-85, carbon-14, and iodine-129. The krypton will be concentrated and stored as a compressed gas in cylinders and the carbon-14 and iodine-129 will be adsorbed and packaged as calcium carbonate and silver zeolite beds, respectively.

Krypton-85, a chemically inert gas, in the packaged form would be a concentrated radioactive source. The dose rate at the surface of an unshielded cylinder would be about 700 R/hr when filled at the treatment plant. Remote operation in shielded storage areas will be required to process krypton, thus reducing the availability of this waste form and making the cylinders relatively inaccessible targets. In case of a release, the material rapidly disperses and the threat to the health of the general public is insignificant. However, a cylinder rupture outside the facility would probably result in serious exposure to nearby operating personnel. The massive shielding required during transport provides protection against sabotage.

Neither carbon-14 packaged as $CaCO_3$ nor iodine-129 packaged as a spent silver zeolite bed are attractive targets for theft and eventual dispersal, or for deliberate dispersal onsite by sabotage. In these forms the carbon and iodine are nonvolatile and nonhazardous in the amounts handled or treated in the facility, and too low in concentration to be a health hazard to the public if released onsite as a result of sabotage.

4.10.2.2 Safeguards Requirements for Storage of Reprocessing Cycle Wastes

During the period before ultimate disposal, solidified HLW may be stored in water basins or in surface facilities in sealed casks. Although the waste is not a source of fissionable material, physical protection during storage must be provided to deter and prevent theft or sabotage. The rationale for either theft or sabotage may be to disperse or threaten to disperse the radioactive contents of the casks or storage facilities.

The physical protection requirements for storage of encapsulated solidified HLW in water basins are expected to be the same as those for storage of spent fuel. If the waste were stored at a reprocessing plant, the facility would be a vital area and its physical protection is described in Section 4.10.2.1. If the waste were stored at a separate water basin facility, the safeguards evaluation for spent fuel storage, described in Section 4.10.1.2, apply. The risk associated with the possible sabotage of solidified HLW in water basin storage is probably less than for spent fuel.

If HLW is stored in a sealed-cask storage facility, the facility would be protected against unauthorized entry, forced intrusion, and sabotage. In such a facility the waste canisters are not readily accessible because:

- Remote operation is required to handle canisters.
- Massive biological shielding is required to attenuate canister radiation.
- Facility design features that protect against severe natural occurrences minimize accessibility of the unloading/handling areas.[a]

The consequences to the public even if a sabotage effort should succeed are, however, expected to be very small. If a canister of waste is ruptured by explosives, the dispersed radioactive material should be confined largely to the storage area because the material is in a solid form and not dispersable except to the extent that pulverization occurs from the explosion energy. Safety analyses of an accidental rupture of a HLW canister inside the building showed that the release of radioactive material would be slight and the public exposure negligible (See DOE/ET-0028, p. 5.4.17).

Packaged TRU wastes are not attractive targets for theft or sabotage because of the low quantities of plutonium and the variable amounts of fission products present. The wastes contain radioactive materials in concentrations several orders of magnitude below those in spent fuel. Much of it would be packaged in 55-gal drums or large boxes. The variations in fission product content will result in surface dose rates that are expected to vary from below 0.2 R/hr to above 10 R/hr.

A sabotage threat will create concerns over radioactive releases. While sabotage may potentially result in some releases to the atmosphere, the amounts released would result in no significant health threats to the public. If a sabotage act causes a bitumen fire, about 10 grams of the fixed waste may be released to the cell, vault or burial crypt atmosphere; lesser amounts would be released to the environment. While an attempted sabotage of TRU waste storage that results in a fire could be a serious incident, the consequences to the public would be small.

The overall physical security required at sites containing TRU wastes protects the public from willful misuse of this waste.

A krypton storage facility will probably be located adjacent to the reprocessing plant. Physical protection of transportation and the storage facility to deter and prevent

(a) Only conceptual plans for such a facility have been prepared. The actual design will involve detailed safety and safeguards analyses.

intrusion or sabotage would be required. The dose rate at the surface of an unshielded krypton cylinder would be about 700 R/hr when received from the reprocessing plant. A remote and shielded storage area will be required for storage, thus reducing the availability of this waste form and making the cylinders relatively inaccessible targets.

It is possible for acts of sabotage to rupture a cylinder of krypton during the receiving or internal transfer operations. The consequences to the public from such acts, however, would be small because the storage buildings are designed to allow the release of krypton through high stacks only. Approximately 104 KCi of krypton-85 might be released over a half-hour period. Such a release could result in significant exposure of workers in the vicinity of the rupture.

Successful sabotage of the krypton storage cell does not appear credible. The cell walls, at least two feet thick, are built of reinforced concrete. However, if the walls are breached by an act of sabotage and krypton is released at ground level, the consequences to workers in the immediate area could be serious but the consequences to the public would be small.

4.10.2.3 Safeguards Requirements for Transport of Reprocessing Cycle Wastes

Solidified high-level wastes will be shipped in casks designed specifically for this purpose. A shipment of HLW will contain more fission product activity but less than 1% of the plutonium included in a shipment of spent fuel. Physical protection requirements for shipments of solidified high-level wastes have not been established by the regulatory agencies. The actual level of physical protection required for shipments of solidified high-level wastes will likely be based on the experience of successful shipments of spent fuel.

Shipping casks as currently conceived, with designs based on the cask criteria for shipping spent fuel, offer significant protection against assault and attempted removal of the contents. A cask would weigh about 90 metric tons, with a special cask cover weighing about one metric ton and requiring special equipment to remove. The cask would be resistant to small arms fire. Explosives in sophisticated designs and arrangements could penetrate a cask. However, the consequences of penetration of a cask would be a minor release of radioactive material at the site (DOE-ET-0029, Vol. 2, p. 8.1.5).

The packaged TRU wastes would be relatively unattractive to an adversary because of its high and varied dilution of radioactive materials. No container or single shipment would contain more than 40 grams of plutonium, and the material, when immobilized in concrete or bitumen or in some other non-dispersible form, would not be a threat to the public as a dispersible radioactive contaminant. If sabotage of a shipment occurs, the release of radioactive materials even under severe conditions is expected to be small (DOE-ET-0029, Vol. 2, p. 8.1.5).

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CHAPTER 5

GEOLOGIC DISPOSAL

In this chapter, the concept of a conventionally mined deep underground repository for disposal of spent fuel and/or fuel reprocessing wastes is described. The status of the technology is described as are uncertainties that require resolution and additional information that would improve confidence in the concept. A description of a conceptual repository for spent fuel or for fuel reprocessing wastes is given. An analysis is presented of the environmental impacts associated with construction and operation of repositories in representative media. Several types of failures of repositories in the long term have been hypothesized to assess societal risk. A description of dollar costs of repositories is also presented. The concern for safeguards is reviewed. Finally, the environmental impacts are summarized in terms of the irreversible and irretrievable commitment of resources and in terms of unavoidable adverse impacts.

5.1 DESCRIPTION OF THE GEOLOGIC DISPOSAL CONCEPT

Geologic disposal of radioactive wastes, as used in this Statement, is the disposal of radioactive wastes in conventionally mined repositories deep within the geologic formations of the earth. Included is the concept of multiple barriers to provide a series of independent barriers to the release of radionuclides to the biosphere.

The multiple barriers that could contain nuclear waste in deep mined repositories fall into two categories: 1) geologic or natural barriers and 2) engineered barriers. Geologic barriers are expected to provide isolation of the waste for at least 10,000 years after the waste is emplaced in a repository and probably will provide isolation for millenia thereafter. Engineered barriers are those designed to assure total containment of the waste within the disposal package during an initial period during which most of the intermediate-lived fission products decay. This time period might be as long as 1000 years in which case the radiation levels and heat generation rates of the total waste would drop by factors of approximately 1,000 and 100, respectively. Engineered barriers must be designed to withstand the more severe radiation and thermal conditions encountered initially.

Two important components of the geologic barrier to be considered in siting are the host rock itself and the geologic surroundings. Properly chosen rock structures provide physical and chemical properties that contribute to repository strength. Sufficient repository depth and lateral extent of the rock mass contribute to the isolation capability of the repository. Tectonic stability and a noncommunicating hydrologic regime combine with rock properties to maintain repository strength and isolation integrity. The geologic barriers can be selected through the site-selection process to provide a stable long-term environment for the waste that is not likely to be disturbed by natural events or human activities.

This section provides an overview of general considerations in the design and location of geologic repositories. Additional details including references to specific studies in the literature are given in Appendix B of Volume 2. Details of both engineered and natural barriers to waste release are also presented.

5.1.1 Factors Relevant to Geologic Disposal

Six factors relevant to geologic disposal are:

1. Depth of repository below the land surface. Presently it is assumed that a range of from 600 to 1,000 m of earth material will exist between the repository and the land surface. This will provide a barrier between the waste and the biosphere and protect the repository from human activities. Dimensions of the host rock are also considered so that the repository will be buffered by rock material laterally and below as well as above it. An artist's conception of a repository is shown together with more familiar structures in Figure 5.1.1.

2. Properties of the host rock. The physical, chemical, and thermal properties of the host rock determine the rock's capability to isolate and contain the waste and reduce unwanted interactions between the rock and waste. These possible interactions include radiation effects on the rock and chemical and physicochemical interactions. Important rock characteristics include strength, permeability, thermal conductivity and expansion, and radiation resistance.

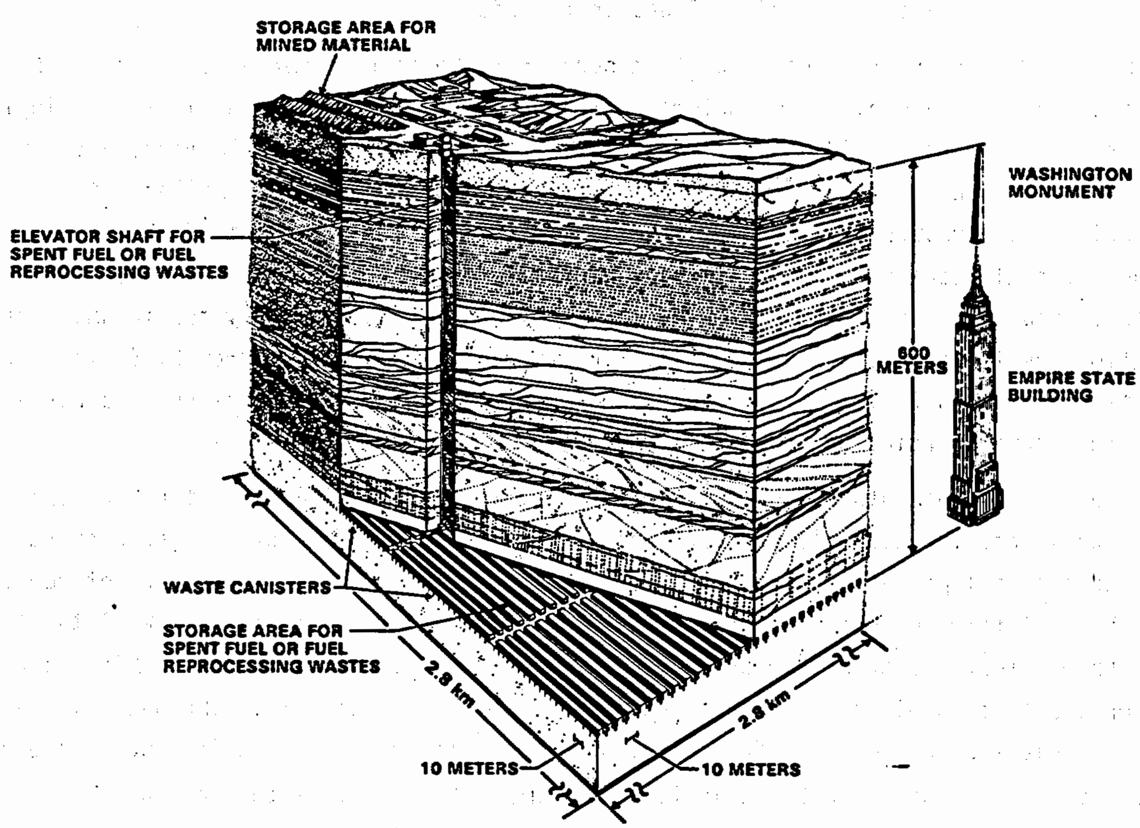

FIGURE 5.1.1. Deep Underground Geologic Waste Repository

3. Tectonic stability of the repository area and region. Proper consideration of this important factor will reduce the likelihood of deformation or disruption of the host rock and thus increase the probability of repository integrity.

4. Hydrologic regime (i.e., surface-water and ground-water considerations). This is important because the existence of connected water channels could provide potential pathways for waste transport away from the repository.

5. Resource potential of the repository site and area. A low resource potential is desirable to avoid loss of any economic resource by the repository existence and to reduce the likelihood of future exploration activities for resource recovery.

6. The multibarrier safety feature. This combines the redundant isolation features provided by the rock properties, the geologic setting, and engineered barriers to give overall added confidence that the waste will remain isolated.

These six factors are discussed in the following sections.

5.1.1.1 Disposal Media Properties

Four geologic media are examined in this Statement to illustrate a range of rock properties for a radioactive waste repository: salt deposits (bedded and dome), granite, shale and basalt. All four rock types possess properties that are favorable for waste isolation. These, as well as some unfavorable characteristics, are discussed in the following pages.

For the purpose of this Statement, the physical properties of a disposal medium describe the characteristics of both the host rock and surrounding rock mass. The disposal rock material is characterized in terms of its texture, i.e., the size, shape, and arrangement of the component crystal grains. Texture is a consideration in the assessment of a medium's behavior under stress and heat, and its hydrologic flow potential.

Rock mass structures include the discontinuities of bedding and joints. Bedding refers to variations in texture because of changes in the sedimentation process by which the rock was formed. It may be present in both sedimentary and metamorph.c rocks. Joints are fractures along which little or no displacement of the rocks has occurred. They are generally formed by extensional release of confining earth pressures. Descriptive features of these discontinuities include orientation, width, spacing, filling material, waviness, and extent (length). The potential for the transport of waste material correlates with the number and extent of host rock discontinuities.

The rock properties of principal interest for waste disposal are those related to strength, stress-strain, thermal, and hydrologic characteristics. These properties and characteristics are discussed and presented in tabular form in Appendix B. For comparative purposes index properties defined as unit weight and natural moisture content are included in the tabulation.

Substantial strength is desirable for engineering design of subsurface repository facilities, especially in maintaining tunnel integrity. Strength properties provide the durability or resistance of a material to processes such as erosion and weathering and breakdown into component minerals. In general, the greater the strength, the greater the ability to resist weathering. Parameters representative of strength include cohesion or friction angle, uniaxial compressive strength, and tensile strength.

Stress-strain properties indicate the deformation characteristics that a material will exhibit under stress. Parameters that describe the nature of the deformation of a disposal medium include Young's modulus, Poisson's ratio, bulk modulus, and shear modulus. These parameters are significant in the analysis of an earth material's strength, durability, and use properties, such as mineability, for isolation. The desirability of (or, trade-offs between) a highly deformable medium versus a rigid disposal medium for isolation purposes is unresolved. The ability of an earth material to deform and seal discontinuities to fluid flow is desirable. Conversely, a rigid earth material is important to the stability of the repository tunnel opening.

Thermal properties indicate an earth material's ability to absorb and conduct heat away from radioactive waste. Knowledge of these properties will allow the evaluation of the effect of the heat upon the integrity of the disposal medium. Pertinent thermal parameters are coefficient of linear thermal expansion, heat capacity, and thermal conductivity. Heat can physically alter an earth material by causing expansion, which, in a confined disposal medium, can jeopardize isolation. For example, too much expansion of the rock might fracture the overburden above the repository. The degree of expansion is dependent on the ability of a host rock to dissipate heat and dependent on the amount of expansion for a given temperature change.

Hydrologic properties are essential to assessing the potential for fluid flow. They are evaluated by the parameters of permeability, hydraulic gradient, and porosity. Restriction of transport of radionuclides requires as low a permeability as possible.

A host rock is an aggregate, composed of one or more naturally occurring minerals and chemical compounds. The constituents provide the chemicals for potential reactions of the host rock with the waste material. These possible reactions may increase isolation by precipitation of insoluble materials or decrease it by converting radioactive waste into soluble compounds. Possible chemical reactions among disposal media, intergranular fluids, and waste must be defined and evaluated for their effect on isolation.

Disposal media of salt, granite, shale, and basalt are examined here and represent only a selected sample of candidate host rock types. Other host rock types may also meet the requirements for media properties and distribution. Additional media can be grouped as having properties similar to those of the example media. Associated disposal media are grouped as 1) salt: anhydrite, gypsum; 2) granite: general crystalline rock, granodiorite, periodotite, gneiss, syenite; 3) shale: general argillaceous rock, carbonate; and 4) basalt: gabbro and some tuffs.

5.1.1.2 Generic Basis for Repository Site Selection

This section presents the generic basis for repository site selection and the design of the repository. Characteristics most desirable for site selection and how they relate to design are discussed. Criteria necessary for development of siting criteria and repository waste form design are presented.

The most important site-selection factors can be derived from the six geologic considerations given in Section 5.1.1.1. In general, the most important factors are the hydrologic regime, the tectonic regime, the multibarrier concept, and the thermal, physical and geochemical properties of the host rock. For any particular location, site-specific considerations peculiar to that site might be different and would take precedence.

The site-selection process will proceed in stages as described below. Program scientists will select regions, areas, and sites, in that order, by their meeting defined requirements. Each stage of the site-selection process will add to the geologic information available for the preceeding stage and will better define uncertainties. Therefore, the site-selection process will yield progressively more significant data; that is, each phase of the process will further characterize site-specific considerations, thus reducing uncertainties.

The following criteria are suggested for repository site selection to assure that the natural barriers function as planned:

1. The repository site shall be located in a geologic environment with geometry adequate for repository placement.
2. The repository site shall have geologic characteristics compatible with waste isolation.
3. The repository site shall have subsurface hydrologic and geochemical characteristics compatible with waste isolation.
4. The repository site shall be located so that the surficial hydrologic system, both during anticipated climatic cycles and during extreme natural phenomena, shall not cause unacceptable adverse impact on repository performance.
5. The repository site shall be located in a geologic setting that is known to have been stable or free from major disturbances such as faulting, deformation and volcanic activity for long time periods.
6. The repository site shall be located in an area that does not contain desirable or needed mineral resources, or to the extent presently determinable, resources that may become valuable in the future.

Regional studies of stratigraphy and structural geology will be conducted to aid in site selection. Stratigraphy is the general characterization of the sequence of rock types both vertically and laterally. Structural geologic studies determine orientation of the rock units in space, direction of dip, configuration of folds, and the characteristics and attitudes of faults, joints, and other discontinuities. Adequate description of the geologic setting may require extensive geologic mapping, some field exploration, and remote sensing surveys, especially in areas that are not yet thoroughly studied.

5.1.1.3 Generic Basis for Repository Design

Several conceptual designs for repositories have been proposed. The surface structures of a repository do not present unique engineering problems. The typical conceptual design of the underground portion of a repository consists of numerous excavated storage rooms (at one or more levels) interconnected by tunnels which serve as transportation and ventilation corridors. The undisturbed rock masses that separate the storage rooms are called pillars. Boreholes will be drilled into the floors (and possibly walls) of the rooms. The waste will be placed in these boreholes. The repository levels are reached from the surface handling facilities through vertical shafts.

The integrity of a repository will depend largely on the state of stress level in the rock, the ground-water flow, the strength of the rock, heating and radiation effects from the wastes, and the layout of the excavations and the disposition of the waste within them. A large body of pertinent data exists which presents and analyzes each of the above factors. The results indicate that there are no fundamental geological or mechanical reasons why excavated repositories should not be used at suitable sites in rock.

The cost of excavating the repository and the cost of rock support systems depend on several interrelated geologic factors: rock strength, rock fractures, rock hardness, rock permeability, rock heating by decay of radioactive nuclides, the state of rock stress, the depth of waste placement, and others. The extent to which these factors influence cost is difficult to determine in advance of construction; unforeseen rock conditions are often encountered in conventional mining operations and in some cases can significantly change the design and the predicted cost. Cost estimates for geologic repositories are given in Section 5.6 of this Statement.

5.1.2 Engineered Barriers

The multiple barrier concept of waste isolation and containment includes both natural or geologic and engineered barriers. Various aspects of engineered barriers are discussed in this section.

5.1.2.1 Engineered Barriers--Waste Package System

The term "waste package" as used in this Statement includes everything that is placed in the waste emplacement hole, e.g., the solidified waste form, canister, overpack, filler and backfill materials, and hole sleeve. The function of the waste package is to:

- Contain the waste for periods sufficient to allow most of the fission products to decay to very low levels.
- Limit the rate of release of radionuclides to the near-field (within the repository proper, see p. K.4) host rock system.

- Limit access of water to the waste and thereby prevent or minimize waste/rock/ leachant interactions.

The functions and materials use for waste package components can be tailored to specific site needs and environmental factors. A conceptual representation of a waste package system is given in Figure 5.1.2. Not all of the components shown here would necessarily be used in all circumstances; the figure illustrates the different kinds of barriers that can be engineered into the waste package. The overlying principle is to design into the package as much redundancy as required by characteristics of the waste to be contained and the characteristics of the natural geologic system.

Waste Package Functions

One may envision how the waste package is designed to function by considering the case of ground water intruding into the repository. A basis for repository site selection will be remoteness from aquifers, so the amount of water should be small. If water intrudes into the repository it would first encounter the backfill, which can be designed to be relatively impermeable to water by reason of its physical and chemical properties. Any water passing through the backfill would encounter a sleeve or overpack, or both, made of corrosion-resistant materials. As a further redundant measure, the canister itself would act as a physical barrier. If all these sequential barriers to water influx were to fail, the waste form itself would be a barrier because of its low solubility and resistance to leaching. If some nuclides were mobilized by ground water, they then would have to travel through damaged package components until they reached the backfill again. The backfill may then function as a sorptive barrier to retard or minimize transport of selected nuclides. Thus, the total waste package system can be designed to minimize the nuclide inventory entering the natural system, by chemically and physically limiting nuclide mobility and by delaying

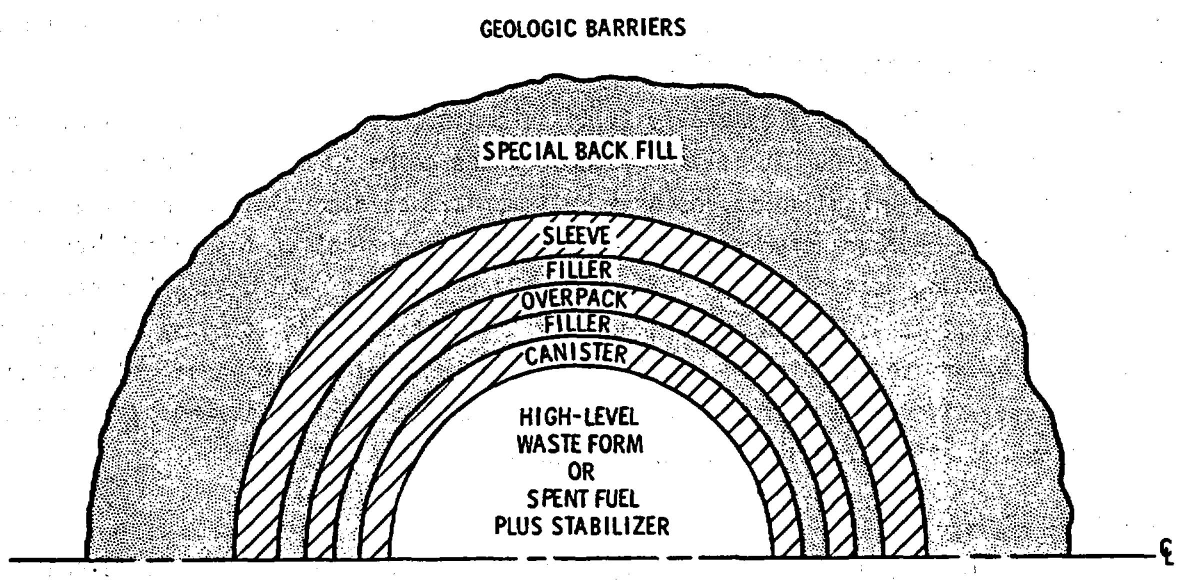

FIGURE 5.1.2. Conceptual Waste Package

releases so that substantial nuclide decay occurs before entering the geologic system where the natural barriers would prevent or delay releases to the biosphere.

5.1.2.2 Waste Packages Components

Components of a generalized waste package were shown in Figure 5.1.2. The following discussion addresses each component separately; however, it is the performance of the entire system of components taken as a whole that is of most importance in the final analysis.

Waste Form

The waste forms include all radioactive materials that may potentially be sent to deep geologic repositories, and are divided into three major categories: spent fuel, high-level waste and TRU waste forms, which are described in more detail in Sections 4.3.2, 4.3.3, and 4.3.4, respectively. The current primary emphasis on waste package design is for spent fuel and for HLW, the reference waste forms considered throughout the following discussion. Due to their high radiation levels and heat generation, spent fuel and HLW place the most stringent requirements on the waste package. However, when most of the fission products have decayed (after a few hundred years), the properties of the TRU waste become dominant.

The waste form is an inert solid designed to be chemically, thermally and radiolytically stable. The waste form itself is the first containment barrier for the waste.

Canister

The canister provides physical containment for the waste forms and thus isolates the waste from near-field surroundings. The extent to which the canister can delay or minimize waste-water interactions is important. Moreover, the canister is expected to provide physical protection during interim storage, transportation, handling, emplacement, and any waste retrieval operations that may be required. The canister material chosen must be compatible with the waste form. The ductility, weldability and impact resistance of metals make them primary candidates as canister materials.

High-level waste forms will generally fill the canister 80 to 90% full. The remaining space will be occupied by air. Stabilizer materials are being considered for use in spent fuel canisters. Gaseous stabilizers, such as helium, have been considered from the standpoint of providing a heat transfer medium without causing chemical or mechanical attack on the spent fuel/cladding assembly or the canister. Particulate or solid stabilizers, such as lead, glass, clay, or sand, can provide additional functions, including maintaining the position of the spent fuel within the canister; preventing canister collapse under lithostatic pressures; acting as a corrosion resistant protective barrier; improving heat transfer; increasing radiation attenuation; and enhancing nuclide sorption.

Overpack

The overpack is similar in principle to the canister. An overpack offers several options to the package designer: it may function as a redundant canister, applied (if necessary) for all stages of package handling, transportation, and emplacement; it can exhibit corrosion or mechanical properties superior to those of the primary canister, thereby providing all, or a major part, of the resistance to the environment required by the package longevity criterion; it can provide a degree of uniformity to a variety of canister types, applied at the repository to accommodate acceptance criteria. The canister and overpack together can be referred to as the "container."

Overpacks for use in the repository are designed especially for chemical durability, with less emphasis on properties such as impact resistance that are mainly important during handling and transportation. Thus, a wide range of materials, in addition to metals, are being studied. These include various ceramic materials, graphite and carbon materials, a wide variety of glasses and specially selected cements.

Emplacement Hole Backfill

Backfill materials are designed to fulfill one or more of several functions:

- Sorbing the limited amount of water that may be present in a repository rock, e.g., from brine inclusion migration in salt.
- Impeding the movement of intruding ground water to and from the waste package.
- Selectively sorbing radioiostopes from ground water in the event of the canister breach.
- Modifying ground-water chemistry and composition (e.g., pH, Eh, etc.) to reduce corrosion rates or minimize waste form leaching.
- Providing mechanical relief by accommodating stresses on the waste package induced by rock movement.
- Serving as a heat transfer medium.

Several layers of filler or backfill material can be utilized, if desired, as shown in Figure 5.1.2; thus, different materials specially designed for specific purposes can be included for optimum functioning of the overall waste package system. Most of the filler or backfill materials being considered are naturally occurring clays, sand or crushed rock that are readily available in large quantities.

In addition to backfill in the emplacement holes, backfill material is also placed in rooms and corridors when the repository is closed. The room and corridor backfill, depending upon the material and method of emplacement, can perform the same functions described for the hole backfill. The degree of structural support provided may be important in preserving repository integrity by limiting the subsidence of room and corridor ceilings. The permeability and porosity of the backfill material may affect the amount of water entering the repository and the time it takes for the repository to become saturated.

Mechanically emplaced crushed rock is used for backfilling the conceptual repository described in this Statement. The use of an engineered sorption barrier as backfill is discussed in Appendix K. Other backfill materials and methods of emplacement are discussed in NUREG/CR-0496 (NRC 1979).

Hole Sleeve

The function of hole sleeves is to maintain open emplacement holes in the repository floor for easy package insertion and retrieval. This may be important if the geologic medium is plastic, e.g., salt or certain shales. In some cases the sleeves could function simply as barriers that, because of their size and bulk, are more easily constructed in situ than transported and emplaced with the waste canisters. Examples of sleeve configurations include cast iron caissons, massive shells of special cements cast in place, or impervious graphite vessels specially bedded in the surrounding rock.

5.1.2.3 Waste Package Development and Assessment

Although most of the ideas incorporated in the multibarriered waste package just described are not new, wide-spread acceptance of the waste package concept is a relatively recent development. A study done in Sweden between 1976 and 1978 did a great deal to promote acceptance of the concept.

The Swedish Approach to Waste Package Design

In April 1977 the Swedish Parliament passed a law which stipulated that new nuclear power units could not be put into operation unless the owners were able to show that the waste problem was solved in a completely safe way. In anticipation of Parliament's action the Swedish power industry formed the Nuclear Fuel Safety Project (KBS) in December 1976 to prepare a response to the government (KBS 1978). A primary objective of the KBS project was to demonstrate how high-level waste or spent fuel can be handled and finally isolated. The study met this primary objective, and although the results were directed to the specific needs of one country and assumed a repository located in granite since that type of rock is widely available in Sweden, the conclusions about the expected performance of the waste packages can have a wider application.

The KBS decided to place reliance on containment for periods of 1,000 yr and 10,000 yr for HLW and spent fuel, respectively; thus, design of the waste package received heavy emphasis. More durable containment for the spent fuel was sought because it produces significant amounts of heat for a longer time than does HLW.

In the proposed Swedish waste management scheme for HLW, the fuel is reprocessed 2 to 10 yr after it is taken from the reactor (KBS 1978, pp. 30-34). The HLW is then vitrified and is placed in cylindrical stainless steel canisters that are stored at the reprocessing plant for at least 10 yr. After this initial storage period, the canisters are shipped to an underground air-cooled dry storage facility in Sweden, where they remain for about 30 yr. Then the packages are prepared for disposal by encapsulation in 6-mm-thick titanium overpacks. To reduce the intensity of radiation emanating from the packages and hence the radiolytic decomposition of the ground water eventually expected to surround the package, a 10-cm-thick layer of lead is placed between the steel canister and the titanium overpack. The packages, now ready for disposal, would be placed in holes approximately 1 m in diameter and 5 m deep in the floors of tunnels in a granite repository approximately 500 m below the surface of the ground. Backfill consisting of a mixture of quartz sand and bentonite is packed around each package. After all holes are filled, the entire tunnel system is filled with a mixture of sand and bentonite similar to that used in the storage holes.

A "reference group" made up of members of the Swedish Corrosion Research Institute concluded that the stainless steel/lead/titanium composite canister could be expected to remain intact for 500 to 1000 yr, even when very pessimistic assumptions were used (KBS 1978, p. 110).

At least two waste package designs appear capable of achieving the longer life sought for spent fuel disposal. In one design the spent fuel is encapsulated, after about 40 yr of interim storage, in copper canisters 77 cm in diameter with walls 20 cm thick (KBS 1978). The other design utilizes a synthetic corundum (Al_2O_3) canister. A feasibility study has shown that it is possible to manufacture such canisters using hot isostatic pressing. Each canister would have an interior diameter of 0.3 m, a 100-mm-thick wall, and be about 3 m long.

Although the Swedish waste disposal packages may be more complex than some packages now under study, they have served to increase our understanding of long-term package performance.

5.1.2.4 Current Status of Waste Package Development in U.S.

Extensive testing and development studies on various individual barrier components of the waste package system, under expected conditions of geologic isolation, have been in progress for several years. These studies are being conducted in industrial and national laboratories and in universities. While most of the studies are not complete, data and results generated during the past few years indicate that components of the waste package system, individually and in combination, can prevent or minimize release of radionuclides outside of the repository by functioning as effective chemical and physical barriers (Katayama 1979, Ross and Mendel 1979, Braithwaite and Molecke 1978, McCarthy et al. 1979, Magnani and Braithwaite 1978 and Nowak 1979).

Through laboratory materials performance evaluation under realistic repository environmental conditions and accelerated aging tests, a number of waste package candidate materials are being selected. Following laboratory testing, nonradioactive bench-scale experiments and radioactive hot cell experiments are planned. These tests employ small-scale mockups of complex systems or groups of system components to investigate the influence of components upon each other. For example, leaching/corrosion studies utilizing a scaled down canister of an actual waste form with rocks and ground waters are in progress (ONWI-9(4)).

The logical culmination of a series of studies investigating waste package material performance and qualification is a field test specific to each repository rock type which involves all components of the waste package. The extent of field testing will be determined from the analysis of earlier results. Various aspects of required laboratory and field tests have been described by the U.S. Geological Survey and the DOE in the Earth Science Technical Plan (DOE/USGS 1980). A Waste Package Design, Development, and Test Plan is being formulated to direct development efforts in an effective and timely manner. An integral part of this plan is the development of coordination among and standards to be followed by researchers and waste management program entities with respect to testing procedures and materials certification. Review and integration entities are defined to include a Materials Steering Committee, a Materials Review Board, a Materials Characterization Center, and an Independent Measurement Standards Laboratory (Hindman 1980). This organization and plan will help assure that waste package design, development and testing programs will produce suitable packages that meet established requirements.

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5.2 STATUS OF TECHNOLOGY AND R&D

Research and development is underway to address the data needs of waste isolation identified in this Statement. In conducting R&D for waste isolation, a technically conservative systems approach is being used, with emphasis on scientific peer review of the activities along with public review, such as the public comment activities of this Statement.

An important document supporting DOE's R&D effort is the Earth Science Technical Plan (ESTP) for Disposal of Radioactive Waste in a Mined Repository (DOE/USGS 1980). The ESTP was prepared by a group consisting of scientists from USGS, DOE and DOE contractors. This group has comprehensively reviewed R&D to define work that may improve the reliability of isolating nuclear wastes in a mined geologic repository, and has recommended programmatic activities. The ESTP describes R&D programs sponsored by DOE and the U.S. Geological Survey. The work in progress involves 76 R&D contractors (including 20 universities and 7 national laboratories). While the key work in progress is discussed in the paragraphs below, the reader is referred to the ESTP to gain more complete perspective on the ongoing R&D activity. Parallel studies sponsored by NRC, EPA and the utility industry are in progress in the United States and in foreign countries (particularly Sweden, Federal Republic of Germany, France, Great Britain, Japan and Russia).

The following sections provide a general discussion of the current status of technology and the R&D activity and requirements for the geologic site selection, waste package, and repository system.

5.2.1 Geologic Site Selection[a]

Geologic site selection involves characterizing promising areas of the United States as possible locations for repository facilities for radioactive waste (see also Section 2.3). During site selection or qualification, certain factors or criteria necessary for adequate performance of the natural system must be considered. Such factors or requirements are summarized in the "NWTS Criteria for the Geologic Disposal of Nuclear Wastes: Site - Qualification Criteria (ONWI-33(2), 1980)." These requirements are being used by DOE to guide its site selection or qualification activities until such time as formal licensing criteria are adopted by the Nuclear Regulatory Commission and the Environmental Protection Agency.

Much of the data base for site selection is available. These data include topography, records of seismic activity and volcanism, hydrology, and presence of the natural resources. Other data, including depth to a potential emplacement zone, areal extent of rock type, attitude (dip, inclination), and the nature of the contiguous formations are developed at specific sites. Ground water, as the principal agent for transport of radioactive waste to the biosphere, has received intensive study and research over the past decade. The principles that govern its occurrence, movement and related rates of supply and usage are well established. While major aquifers and their distribution and properties are known, additional study using accepted techniques can define regional and local flow systems adequately.

(a) Section 2.3 describes the present National Site Characterization and Selection Plan.

Specific topics elaborating on site selection criteria and the supporting R&D addressing these matters are discussed below. Supporting R&D projects are listed by organization in Appendix L of Volume 2.

5.2.1.1 Methods for Regional Geologic Studies

Geologic studies will identify, for a specific region, area and site, the current state of stability and the geologic processes which have acted in the past. Based on this information along with repository design, the projection and probability for the future stability of the specific site will be estimated (see Appendix L).

General geologic conditions in the United States are well known and have been extensively described (Geologic Society of America, current listings). Exploration for mineral resources--notably oil, gas, coal, and metals--by private industry provides much information about sub-surface geologic conditions, in many instances to depths approaching 10,000 m (Am. Assoc. of Petroleum Geologists, current listing). The construction of nuclear reactors, which must meet stringent licensing requirements, has resulted in detailed geologic evaluations of areas in the Eastern, Midwestern, and Far Western United States (FUGRO, Inc., 1977). Moreover, various universities have developed as centers of detailed geologic information on specific subjects. The accumulated knowledge is sufficient to identify areas in the United States that meet many of the requirements (Section 5.2.1) for radioactive waste repositories.

5.2.1.2 Methods for Site Analysis

In general, geologic studies are the mechanism by which available data about the subsurface environment are synthesized and coordinated to assess whether the stratigraphic and structural settings of a proposed site are suitable for a waste repository. Remote sensing and geophysical studies are conducted to support this activity. Geologic interpretations are the basis for defining models by which the hydrologic, geologic, geochemical, thermal, and mechanical characteristics of a repository are assessed.

Geophysical Surveys

Geophysical surveys are an integral part of site selection and characterization studies. Many of the geophysical techniques utilized by the petroleum and mineral industries have been applied to the search for geologic repositories. The broad categories of exploration geophysics summarized in this subsection are gravity, magnetic, electrical, and seismic methods. In addition, well logging and borehole geophysics are discussed.

There currently exists a wide variety of geophysical techniques available for site selection and characterization. Geophysical surveys are a well established part of exploration prospecting and proper evaluation can provide extensive information about subsurface geologic conditions. Such surveys are especially valuable in repository investigations because they permit investigation of subsurface conditions without extensive drilling.

Gravity analysis can detect small variations in the earth's gravity field (Dobrin 1960). The variations of principal interest to repository siting result from lateral variations in subsurface rock density. Density variations may result from deformed strata, faults, igneous intrusives, diapirs, breccia pipes, or lithologic changes.

Magnetic methods detect variations in the earth's magnetic field (Dobrin 1960).The magnetic variations (anomalies) of interest to site studies are due to lateral changes in mineral content (especially magnetite) or to variations in the remnant magnetism of igneous rocks. Subsurface structures like anticlines or faults can be detected if they result in lateral changes of the above properties (Fabiano 1976).

A variety of electrical methods (Dobrin 1960 and Keller 1966) are used in geophysical exploration; all depend upon detecting variations in the electrical resistivity of the media through which a current flows. Subsurface resistivity is highly variable and strongly influenced by the amount and the nature of fluids in the rocks. For this reason, such hydrologic features as dissolution of salt, ground-water tables, and porosity variations are particularly amenable to electrical prospecting methods.

Seismic exploration methods are perhaps the most useful geophysical tools for obtaining accurate representations of the subsurface geology at individual sites (Dobrin 1960). They rely on the reflection or refraction of seismic (acoustic) signals due to contrasts in velocity or acoustic impedance (the product of seismic velocity and rock density). Acoustic signals are usually introduced into the earth by explosive sources or vibrating or impacting masses. Seismic reflection surveys are particularly useful in mapping the attitude and continuity (or lack thereof) of subsurface rock beds. Other methods and equipment utilized for seismic reflection can be selected for the specific site and parameters (i.e., depths, dimensions) of interest. These parameters are often defined to provide information from depths of more than 1,000 meters (Vail et al. 1978). Special field parameters and techniques (high-resolution seismic) are available to explore accurately the shallower depths of interest for repositories.

Geophysical logs in well bores are a powerful tool for correlating and interpreting subsurface geologic conditions, including the condition and fluid content of subsurface rocks (Dobrin 1960). They supplement cores and rock samples and furnish a vertically continuous record of certain physical properties for each borehole. Many types of logs are used. Focused resistivity logs provide a reliable measure of in-situ rock and fluid characteristics. Microresistivity logs measure the properties of small volumes of rock just behind the borehole wall and thus permit the boundaries of permeable and/or electrically resistive formations to be sharply defined. Gamma-ray logs indicate the clay content of various formations and are valuable in making lithologic interpretations in clastic rock sequences. Neutron logs are useful for identifying porous rock strata and rock densities. These logs respond mainly to the hydrogen content of the formation and indicate the presence of water, oil, or hydrogen-bearing minerals. Acoustic logs measure the velocity of sound in rock units and can also help determine the porosity of a formation.

Hydrologic Technology

The role of hydrologic studies in site exploration can be separated into three overlapping areas: 1) two-dimensional characterization of the surface and ground-water systems for the region or hydrologic basin in which the site is located, 2) three-dimensional characterization of ground-water conditions at candidate sites, and 3) the potential effects of the repository, the climate, or other perturbations of the ground-water system.

Because it is believed that hydrologic transport will be the principal mode of translocation of radionuclides, a considerable amount of field and test data will be acquired to assess the hydrologic system. The techniques for obtaining most of the data are currently available; others, including improved techniques for ground-water dating, fracture-flow modeling, and permeability determinations for low permeability rocks, need development (Barr et al, 1978 and Bredehoeft et al. 1978). Hydrologic models combined with geochemical studies are used to estimate the likely composition and concentration of any and all radionuclides at any given point and time relative to a site's regional aquifer system.

Data from hydrologic testing are combined with geologic interpretations of a site and region to produce a detailed three-dimensional model of the near-field (see p. K.4) hydrologic flow system which includes the fracture-flow conditions. This is then integrated with thermal and mechanical models to calculate the near-field disposition of the wastes should they escape containment. The near-field models determine the source terms for regional two-dimensional flow models of a subject hydrologic basin. These regional models are used to calculate the isolation potential of the far-field natural system. Retardation mechanisms (e.g., sorption, precipitation and diffusion into the rock matrix) and radioactive decay chains for the radionuclides will be factored into both near- and far-field models of the isolation system. Conservative assumptions regarding potential changes in the hydrologic system that may be caused by climatic and tectonic changes will be used to develop scenarios for modifying models of present ground-water flow conditions.

Permeability, effective porosity, and rock compressibility can be determined by pump or injection tests in wells at the depth intervals of interest. Hydraulic properties are routinely measured for laboratory specimens of core or other rock samples obtained from the site (Lin 1978). Using appropriately spaced wells, hydraulic communication between them can be established during pump or injection tests (Davis et al. 1966) to provide reliable calculations of in-situ ground-water velocities.

Isotopic dating of ground water (Barr et al. 1978 and Bredehoeft et al. 1978) provides an alternative reference for evaluating calculated velocities. Water can be sampled for dating from selected discharge points and well locations throughout the ground water basin considered likely to be influenced by a repository. Differences in water ages among sampling points are used to calculate natural velocities.

The identification and analysis of hydrologic conditions in nearly impermeable rocks is necessary to establish the degree of impermeability possessed by the host rock unit

(Witherspoon 1977). Pulse injection tests aid in determining permeability in low permeability rocks (Ballou 1979). Moreover, pressure decay curves for gases pressurized at selected borehole intervals can be used to estimate the permeability of the very tight rocks expected at repository horizons. Although present measurement techniques for hydraulic conductivity in nearly impermeable rocks may be in error by up to a few orders of magnitude (Bredehoeft et al. 1978), even the higher, most conservative values indicate that water moves extremely slowly in these rocks.

Hydrologic R&D Studies

For rocks that possess a natural fracture system (e.g., granite, basalts, some shales, limestones, sandstones) the determination of near-field flow mechanisms is also evolving. Because fracture networks are not random, their nature and orientation within the system will be statistically determined. Methods designed to assess fracture effects on hydrologic flow are currently being developed at the Nevada Test Site (Johnstone 1980), the Stripa mine in Sweden (Gale et al. 1979), and the Los Medanos site in New Mexico (Gonzales et al. 1979). The direct determination of hydrologic parameters in fracture networks includes conventional pump testing with multiple-point piezometers, tracer studies, and flow-meter tests performed in wells or subsurface facilities constructed at the repository site or in rock bodies that provide a close analog of site conditions.

Water influx at mines in crystalline rocks is a well-known phenomenon. However, where permeabilities are very low, mine ventilation commonly evaporates and removes most, if not all, of this water (Gale et al. 1979). Thus, the mines are usually "dry," although a small amount of water may continually flow into them. By sealing a room with airtight bulkheads and circulating controlled quantities of warm air, the amount of seepage water can be determined by measuring the humidity and mass of the circulating air. Data on fluid gradients around the sealed-off chamber permit calculations of nearby rock permeabilities. Such an experiment is being performed at the Stripa mine in Sweden (Gale et al. 1979 and Lawrence Berkeley Laboratory 1978).

Site-Specific R&D

The thermal properties of potential host rocks can be measured in the laboratory by accepted methods (Stephens et al. and Jaeger et al. 1979). Standard sized cylindrical specimens are subjected to a controlled thermal power source at one end; increasing temperatures and dimensions are measured either along the axes or along the outside lengths of the specimens. The results are then used to calculate volumetric expansion coefficients and thermal conductivity. The specific heat of a rock is determined by standard calorimetry (Stephens et al.).

Mechanical properties of potential host rocks can also be measured in the laboratory by standard techniques and apparatus (Jaeger et al. 1979); the results are used in preliminary models of the repository's response and to help determine which properties require better definition by field testing (Chan et al. 1980). The compressive strengths of potential host rocks are determined in accordance with well-accepted methods by observing which states of stress and temperature cause fracturing. Standard tests are also performed to determine the tensile strength of rocks.

Synergistic effects between thermal and mechanical properties are determined for laboratory samples by obtaining data on mechanical response as a function of rock temperature or obtaining thermal conductivity data as a function of rock stress. The effects of the rock's fluid content on specific heat, critical stress, and thermal conductivity are also being investigated.

Sorption capacities are currently determined by passing water doped with radionuclides through the rock and measuring the amounts of radionuclides retained. Transient batch methods for determining sorption are currently being standardized (Brandstetter et al. 1979). Techniques are also being developed to identify minerologic and molecular affinities for sorbed radionuclides, allowing a better understanding of the materials and mechanisms responsible for the sorption process.

Laboratory tests are being validated by field determinations of thermal, mechanical, and chemical behavior under expected repository conditions. Field tests generally involve single or multiple heat sources emplaced in drill holes with an array of measuring instruments surrounding the heat source. A monitor array can be designed to measure rock temperatures, deformation, water content, chemistry, and rock stresses as a function of time and distance from the heat source.

Regional Geologic Forecasting Studies. Predicted performance of a geologic system has not matured to the point enjoyed by conventional engineering disciplines. Geologic research has largely concentrated on characterizing present-day natural processes and events and on historically reconstructing the distribution, magnitude, and sequence of past events. However, future tectonic activity, including volcanic eruptions, folding, epeirogeny, fault movements, salt diapirism, and seismic activity, need to be predicted to the degree that the likelihood and the consequences of changes in the natural system with regard to containment and isolation can be estimated.

Plotting space-time relationships of past events allows a calculation of past rates and distributions of occurrence for tectonic events (Crowe 1978 and Rogers et al. 1977). The probabalistic extrapolation of these rates into the future must be weighted against deterministic tectonic models such as plate tectonics to determine whether observed space-time distributions are likely to continue or be modified. The geographic scale for which data are compiled is of critical importance and needs to be evaluated. In general, for larger areas, consensus is more readily obtained among earth scientists about tectonic processes. Conversely, averaging probabalistic projections for individual events over large areas decreases their reliability for a given site. Thus, a reasoned interpretation of probabilistic and deterministic approaches is required to assess the likelihood of tectonic events that might disrupt a repository's natural system. This combination of methods is most developed for assessing seismic hazards (Algermissen 1976 and Glass et al. 1978).

Potentially active faults can be deterministically identified from geologic, geophysical, seismic, and natural stress data. Standard earthquake-hazard assessment provides probabilistic estimates of expected return periods at specific sites for ground motions of various magnitudes. These methods are used in conjunction to help determine appropriate seismic design requirements. Similar methods are evolving for volcanic and diapiric phenomena.

The consequences of tectonic events must also be estimated. Observations of earthquake-related damage, both at surface facilities (Lew et al. 1971) and in mine tunnels (Pratt et al. 1978 and Dowding et al. 1978), provide empirical data for substantiating calculations based on the physical response properties of the structures of interest.

The consequences of such intrusive processes as salt diapirism and volcanism are estimated by studying the geometry, disruption zones, and chemical alterations associated with existing intrusions. Where conditions allow current study, the movement of faults, intrusions of material, and tectonics are evaluated also in terms of their effects on hydrologic systems and erosional processes. Impacts of faulting, erosion, and intrusion are estimated parametrically by assuming various event-scenarios and analyzing their effects on the hydrologic flow models.

The prediction of tectonic events and their potential impacts over periods of tens of thousands of years is an advancing capability. Careful selection of repository sites can reduce the likelihood of tectonically induced disruptive events to almost zero. The potential impacts of postulated events will be defined by scenario analysis in order to assess their effects on containment and isolation.

Resource Studies

The potential for exploiting mineral, energy, water, and subsurface land-use resources both now and in the future will be assessed throughout the site-selection process. Geologic, geophysical, borehole, and geochemical studies conducted during site exploration and qualification provide data for evaluating the potential for resource development. The exploration and ultimate selection of a repository are the converse of seeking an ore body or an oil field, in that investigations are conducted to locate areas with a low resource potential. If any characteristic, including thermal gradients, in the site location significantly exceeds the crustal average, its potential value to future generations needs to be carefully considered. The consequences of inadvertent human intrusion into the repository due to resource exploration at some future time must also be considered.

Status of Ongoing Exploration Programs

Preceding sections have described the factors of the natural system important in site selection, design, and construction of deep geologic repositories; the requirements that must be satisfied by a repository site; and the methods available or being developed for characterizing and assessing the natural system.

This section identifies site-specific geologic investigations conducted over the last several years. The site characterization process, described in Section 5.1.1.2, will be conducted in four steps: national screening surveys, whose objective is to identify places that have some potential for waste isolation; regional studies, which evaluate a specific region of interest; area studies, which are conducted to characterize the areas of interest described by the regional study; and location studies, which further narrow the scope of the investigation to a site or sites.

Individual investigations are in various stages of the site-characterization process. Current investigations include 1) the Gulf Interior Region salt domes, 2) the Paradox Basin, 3) the Permian Basin, 4) the Salina Basin, 5) basalt flows at the DOE's Hanford Site, and 6) DOE's Nevada Test Site. Because of the generic nature of this Statement, details of site-specific studies are not included; for details regarding regional studies, the reader is referred to DOE's position statement to the NRC Confidence Rulemaking (DOE/NE-0007).

5.2.2 Waste Package Systems

Package components consist of the waste form, stabilizer, canister, overpack, sleeve, and backfill (Section 5.1.2).

Testing and development studies on various individual barrier components of the waste package system under expected conditions of geologic isolation have been in progress for several years. These studies have been conducted in industrial and national laboratories, as well as universities, both in this country and abroad. Most of these studies are not complete, but data and results generated during the past few years do indicate that components of the waste package system can prevent or minimize release of radionuclides to the natural system by functioning as effective chemical and physical barriers. Programs, program plans, and results are described in DOE/NE-0007 (DOE 1980).

Because of the many candidate materials for the waste package, package development programs will proceed in a logical sequence of scale and complexity. The following sequence of testing is planned:

- Initial laboratory testing using simulated waste
- Laboratory testing using real waste
- Large-scale testing in the field involving all components of the waste package.

Various aspects of the above tests have been described by the U.S. Geological Survey and DOE in the Earth Science Technical Plan (ONWI 1980), which discusses the types of data required and the sequence of laboratory, large-scale engineering, and field demonstration tests.

5.2.2.1 Waste Form

Presently, DOE has experience with spent fuel and glass as waste forms. In order to determine whether present-day spent fuel can be expected to behave satisfactorily in a geochemical environment, studies are being conducted to determine whether the release rates of waste nuclides are controlled by diffusion from UO_2 when the oxygen content of water is held to very low values (ONWI 1979). To date the information obtained from such experiments indicates that lowering the oxygen content of the water can significantly decrease the release rate of the nuclides. Preliminary results indicate that, although some radionuclides are released more rapidly than others as a function of experimental conditions, spent fuel is a durable waste form that exhibits low release of radionuclides when subjected to ground water under normal repository conditions.

Historically, glass, particularly borosilicate glass, has been the major focus of alternate waste form work, and in 1977 it was selected as the reference material for immobilization of the Savannah River Plant high-level waste (Stone et al. 1979). Small-scale operating facilities have demonstrated practicality of the vitrification process (EPRI 1979). In addition to U.S. work, studies and pilot plants involving glass are under way in France, Germany, Belgium, and England. Recently, however, more attention has been devoted to other waste forms, and studies are being conducted to evaluate their characteristics (DOE 1979).

A number of other waste forms are being studied (ERDA 1976, DOE 1979). Prior to the decision to defer reprocessing, significant progress had been made in the development and testing of waste forms, such as glass, for wastes generated by commercial reactors. Subsequent to that decision, the emphasis of work on alternate waste forms has shifted to defense related wastes. DOE is continuing to sponsor work on alternate forms, and it is fully expected that the results and technology developed would be transferable, in large part, to the commercial waste program and existing liquid wastes (EPRI 1979).

5.2.2.2 Materials

For filler materials as stabilizers in spent fuel canisters, candidate materials include lead, glass, clay, sand, inert gases (e.g., helium) and castable solids (e.g., glass, lead and lead alloys, zinc and zinc alloys) (Jardine 1979 and Morgan 1974). Basic physical and chemical properties of candidate stabilizer materials are well known. Some of these candidate materials have been evaluated (under expected repository conditions) for use as barrier materials other than as stabilizers (e.g., as canister, overpack, and/or backfill barriers). Since the overall waste package functions are similar (e.g., corrosion resistance, nuclide sorptive properties, protection of the waste form), the same materials testing can, in many cases, be applied to several system components.

Canister, Overpack, and Sleeve. Candidate material selection for canister and overpack will be based largely on the results of corrosion tests as a function of temperature, radiation, and ground-water chemistry (e.g., pH, Eh, composition, and ionic strength) that are typical of the water in various media of interest (i.e., basalt, granite, salt, and shale). Applicable materials studies to date include consideration of general corrosion rates, pitting and crevice corrosion susceptibilities, stress corrosion cracking, effects of oxygen concentration, solution volume to solid surface area ratio, and possible effects from radiolysis products (Braithwaite 1979 and Magnani 1979). Filler material may also be used between the canister, overpack and sleeve.

Emplacement Shaft Backfill. Closure of the loaded repository will require backfilling the waste emplacement shaft; backfill materials are being tested for selective nuclide sorption properties (for fission products and actinides), to significantly reduce radionuclide migration through the backfill barriers.[a] The capability of the backfill materials to prevent or delay ground-water flow through the backfill is also being evaluated. Other properties of interest being evaluated (Neretnieks 1977 and Nowak 1979) are thermal conductivity, mechanical support strength, swelling, plastic flow, and forms and methods for emplacements (DOE Statement of Position to NRC (DOE/1980).

5.2.3 Repository System

The repository system will provide for the receipt, inspection, transfer to the underground, emplacement, and containment after closure of radioactive wastes. Performance criteria stipulating the minimum acceptable behavior for an engineered system are required in evaluation of the design. Criteria for the performance of the mined repository during the operational phase have not yet been established; however, such criteria are expected to be similar to those for other nuclear packaging and storage facilities.

The surface facilities of a repository are similar to those now used in the nuclear industry. Radiation protection practices in the repository, therefore, will be similar to those used in other nuclear facilities and are not discussed here. Repository support facilities and underground workings are also similar in many ways to those common to the mining industry. Therefore, issues not uniquely related to radioactive waste repositories, such as the construction of support facilities, are not discussed here.

For the purpose of assessing the long-term containment and isolation integrity of a geologic repository, disruption phenomena which represent potential waste release mechanisms have been postulated. This analysis is discussed in detail in Section 5.5. Existing studies show no compelling environmental reasons, including public health, that should preclude disposal of waste in deep geologic repositories.

Other scenarios and variations of the scenarios presented in this Statement have been analyzed and published (Claiborne and Gera 1974). The conclusions of the published studies are in agreement with those provided above. However, this is a complex and extensive area of ongoing research which is generally being examined by scenario analysis, study of waste form release rate and radionuclide transport phenomena, and consequence analysis. Specific R&D projects in risk assessment are listed in Appendix L.

Discussion of potential adverse impacts of constructing and operating a repository will be limited to the following factors:

- Excavation and underground development
- Thermal effects
- Radiation effects
- Repository penetrations.

(a) Such materials are sometimes referred to as "getters" due to their ability to retard the movement of certain materials.

5.2.3.1 Excavation and Underground Development

The excavation of rooms and tunnels underground will induce a new stress state and displacement field in the host environment. The nature of these stresses and displacement fields depends on the cross-sectional geometry of the excavation, the layout of the tunnels and rooms, and the extraction ratio (the ratio of the volume removed to the volume remaining) (Koplick et al. 1979).

Fracturing around the perimeter of the tunnels and rooms and effect on in-situ stress states and its implications for long-term containment are two potential impacts being considered in the excavation of a repository.

Vast experience has been gained in the excavation of various kinds of underground facilities. Fracturing during drilling and blasting operations is limited by controlling such parameters as the size and type of charge, the configuration of drill holes, and the sequence of detonation. Controls of these types are used extensively in the excavation of undergound facilities intended for storage purposes and for long-term operations (Svanholm et al. 1977); examples are caverns for compressed air and natural gas storage. In-situ tests are in progress to confirm their suitability for the excavation of mined geologic repositories (Hustrulid 1979). It is believed that no further technological advances are needed in this area (Guiffre et al. 1979).

5.2.3.2 Thermal Effects

The thermochemical impacts of principal interest in repository design are those that would accelerate the degradation of the waste package and the migration of radionuclide away from the package. The introduction of heat into the system will change the environment in which the waste was placed. The design of a waste package capable of withstanding the heat-altered environment is discussed in Section 5.1.2.

The introduction of heat into the natural system will induce stresses in the host rock and surrounding media (IRG 1979 and NAS 1979). These stresses will be superimposed on the existing stresses and must be considered in design to ensure structural stability of the repository. The heat generated by the emplaced waste will cause the rock mass to expand, thus inducing surface uplift. In the long term, as the heat generation rate decreases, the surface will subside. Displacement of the overlying rock mass may cause fracturing in the rock, thereby giving rise to perturbations in the hydrologic flow regime. In addition, heat may modify the thermal and mechanical properties of the rock; for example, an increase in temperature will enhance the ductility of a rock but reduce its ultimate strength.

5.2.3.3 Radiation Effects

The effects exerted in the host rock by irradiation have generally been considered to be of secondary importance. To date, most of the laboratory and theoretical studies have concentrated on the effects of radiation on salt. The information available on radiation effects on salt and on other geologic formations of interest for waste disposal has been compiled (Jenks 1975). It is desirable, at this point, to conduct in-situ tests to determine the effects of radiation of interactions between the host rock and the waste package and to ascertain whether deleterious reactions occur due to synergism among the heat, radiation, and chemical interactions with the package (Carter 1979).

5.2.3.4 Repository Penetration

In general, the penetration of host rock by shafts and boreholes will be expected to have small environmental or safety consequences. Consideration of final sealing will require the evaluation of excavation techniques, the effect of excavation on the host rock (fracturing), and changes in rock stresses. Testing of plugging technology for shafts and bore-holes is in progress. Studies planned or under way addressing this matter are listed in Appendix L.

5.2.4 Summary

The following summarizes the present status of technology and R&D in support of improving the reliability of a mined geologic repository.

- The general criteria that have been proposed for repository site qualification have been identified in the "NWTS Criteria for the Geologic Disposal of Nuclear Wastes: Site - Qualification Criteria (ONWI-33(2), 1980)."."
- Studies of the natural geological system, development of the man-made waste package, and repository system analysis will all combine to lead to repository designs that utilize multiple barriers to their maximum efficiency in a repository.
- Regional geologic conditions in the U.S are well known and have been extensively described; geologic forecasting is being accomplished by extrapolating past geologic-event data into the future and weighing results against deterministic tectonic models.
- Ground water as the principal agent for transport of radionuclides to the biosphere has received extensive study and research; the principles that govern its occurrence and movement are well established. Additional studies are being conducted, using accepted techniques, to define regional and local ground-water flow systems.
- Sorption capacities of candidate rock media in contact with radionuclides are being determined in the laboratory. These data are designed to permit estimation of long-term migration of the radionuclides in repository host media.
- Continued development of the waste package is expected; studies with candidate materials for the waste package development will proceed in a logical sequence and scale of complexity.
- The repository system performance will be affected by excavation and underground development, thermal effects, radiation effects, and repository penetrations. These effects are being evaluated individually and synergistically for effects in overall repository performance.

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5.3 DESCRIPTION OF THE CONCEPTUAL GEOLOGIC REPOSITORY FACILITIES

In this section, descriptions are given of a conceptual repository for spent fuel in the once-through cycle and a repository for wastes from the reprocessing cycle. The conceptual repositories are described independent of geologic media with specific design and operational features that may be affected by geology detailed separately. Geologic media considered representative of formations available for location of a repository and that are described in this Statement are bedded salt, granite, shale, and basalt (other media may also be acceptable). The concept of siting repository facilities on a regional basis is also described in this section.

5.3.1 Once-Through Fuel Cycle Repository

Conceptually, a repository operating in the reference once-through fuel cycle is required to receive PWR and BWR spent fuel elements. The characteristics of these wastes are described in Section 4.2.

5.3.1.1 Design Bases

Waste emplacement at the conceptual repository is controlled by thermal criteria. The thermal criteria used here specify both areal thermal loadings, which control canister spacing, and canister thermal loadings, which limit the heat output of individual waste packages. The criteria were developed from an analysis of the thermal stresses that accumulate in the geologic formation and in the waste canisters. The criteria are designed to limit these stresses to values that will not compromise the integrity of the formation, the mine area or the waste canisters. Development of these criteria is discussed in Appendix K.

The design areal thermal loadings for the conceptual repositories for this Statement were limited to two-thirds of the calculated allowable thermal loadings. This was done to ensure a conservative estimate of capacity. These design basis thermal limits for spent fuel are shown in Table 5.3.1.

The criteria for granite and basalt, 320 kilowatts/hectare,[a] indicate that 2.6 times more heat-generating waste may be stored in a hectare of granite or basalt than in a hectare of salt. This means that with equal areas a repository in granite or basalt would contain approximately 2.6 times more spent fuel than a repository in salt. This ratio is actually 2.4 for the conceptual repositories because of differences in the mining extraction ratios and room arrangements between the hard rocks and salt. Another parameter, discussed further in a later subsection, that affects the repository waste capacity is waste age.

We assume here that spent fuel may be sent to a geologic repository after five years of cooling. However, a large portion of the spent fuel will be considerably older and cooler. This is because of the large inventory that will accumulate before a repository is available and because of the time required to dispose of this inventory. For a 1990

(a) One hectare equals approximately 2.47 acres.

TABLE 5.3.1. Conceptual Repository Design Thermal Limits for Spent Fuel

Medium	kW/ha[a]	kW/acre[a]
Salt[b]	124	50
Granite	320	130
Shale	200	80
Basalt	320	130

(a) Area occupied by the emplacement rooms and their associated pillars only.
(b) The placement of spent fuel in salt is limited by long-term surface uplift. The degree of surface uplift is dependent upon the thermal loading averaged over the full emplacement area (corridor area as well as rooms and pillars). Two-thirds of the allowable average thermal loading for spent fuel in salt is 100 kW/ha (40 kW/acre). The thermal loading listed in this table (124 kW/ha) is the room and pillar area loading that results in 100 kW/ha average loading. Room and pillar integrity is the controlling criteria in other rock media and is dependent upon the room and pillar loadings listed in the table.

repository startup, the earliest date considered in this Statement, the average age of spent fuel available for the first repository was calculated to fall within the range of 7 to 11 years. For a later repository startup the spent fuel will initially be much older (See Section 7.3). For the conceptual repository described here we assume that the average age of the spent fuel delivered to the repository is 6.5 years old. The criteria in Table 5.3.1 were developed for 10-year-old fuel. Using those criteria for 6.5-year-old fuel provides an additional degree of conservatism since the thermal limit tends to increase for younger waste. There are also thermal limits for the individual canisters, but for the spent fuel repository concept used here, where the canisters contain only a single fuel assembly, the thermal output of the canisters is always well below the limit.

In the absence of detailed site-specific geologic data, optimization of the repository design to account for the special qualities of each medium is not possible. Instead a standardized repository design using a conventional underground layout is specified with an overall area of approximately 800 ha (2000 acre). This area provides reasonable waste capacity and is achievable from both construction and operational points of view. Actual repositories may be either larger or smaller than 800 ha depending upon specific site characteristics and more detailed operations analyses.

Repository design, construction, and operations presented here assume a homogeneous geologic formation without major flaws or discontinuities. This simplifying assumption is appropriate for use in this generic analysis; actual repositories will have site-specific design features. The design may involve preparation of a preliminary repository layout on the basis of initial site investigations. The preliminary layout would be modified as construction progresses and the formation is more fully explored.

For the conceptual repository described here, excavation of the full underground repository area is postulated to be completed during the first five years of repository operation. During this period all wastes are emplaced retrievably to allow their timely removal should events during construction warrant this action. The retrievable period also provides an opportunity to evaluate the repository interface with emplaced wastes. Instrumentation will be installed to monitor temperature profiles in the waste and rock and to measure room and pillar stress and deformation. Results of these studies may verify repository design or indicate the need to modify waste emplacement procedures.

5.3.1.2 Facility Description

The conceptual repository consists of 1) surface facilities for waste receiving and handling and for mining and general operations support and 2) subsurface facilities for waste handling and emplacement and for mined rock removal. Surface facilities, shown in Figure 5.3.1, are similar for all repositories regardless of geology. These facilities and the mined rock storage pile constitute the visible evidence of the repository and occupy an area of about 180 ha at the salt and shale repositories and 280 ha at the granite and basalt repositories. The additional 100 ha at the granite and basalt repositories are required for larger amounts of rock that are mined from these formations to accommodate the additional waste disposal capacity resulting from higher thermal limits. Figure 5.3.2 provides an artist's concept of a geologic repository.

All surface structures in which radioactive wastes are handled are operated at less than atmospheric pressure. Ventilation flows are controlled by pressure differential from areas of low contamination potential to areas of successively higher contamination potential. Exhaust air is processed through a roughing filter and two high-efficiency particulate air (HEPA) filter banks in series prior to discharge via the 110 m mine ventilation stack.

Additional details of surface facilities at the repository are found in DOE/ET-0028.

The conceptual repositories for the once-through fuel cycle require three shafts in salt and shale and four shafts in granite and basalt to support waste handling and mining operations. These are the canistered waste (CW) shaft, the men and materials (M&M) shaft, and ventilation exhaust (VE) shaft in all the media and the mine production (MP) shaft in granite and basalt to support the larger mining effort.

The canistered waste shaft provides a means for transporting the canisters of spent fuel from the canistered waste building to the subsurface emplacement areas. The men and materials shaft is provided to handle mine and storage personnel, equipment, ventilation air and mined rock during excavation and backfilling. The ventilation exhaust shaft is divided into two compartments to provide separate exhaust for mining and for placement operations. The shaft discharges into the ventilation exhaust building.

The mine production shaft contains skip hoist equipment for removal of mined rock to the surface and supplies additional ventilation air to the mine.

The repository underground layout is a conventional room and pillar arrangement that serves the need for repository ventilation, opening stability, thermal effects, and efficient use of excavated space. Of the 800 ha underground area, actual spent fuel emplacement

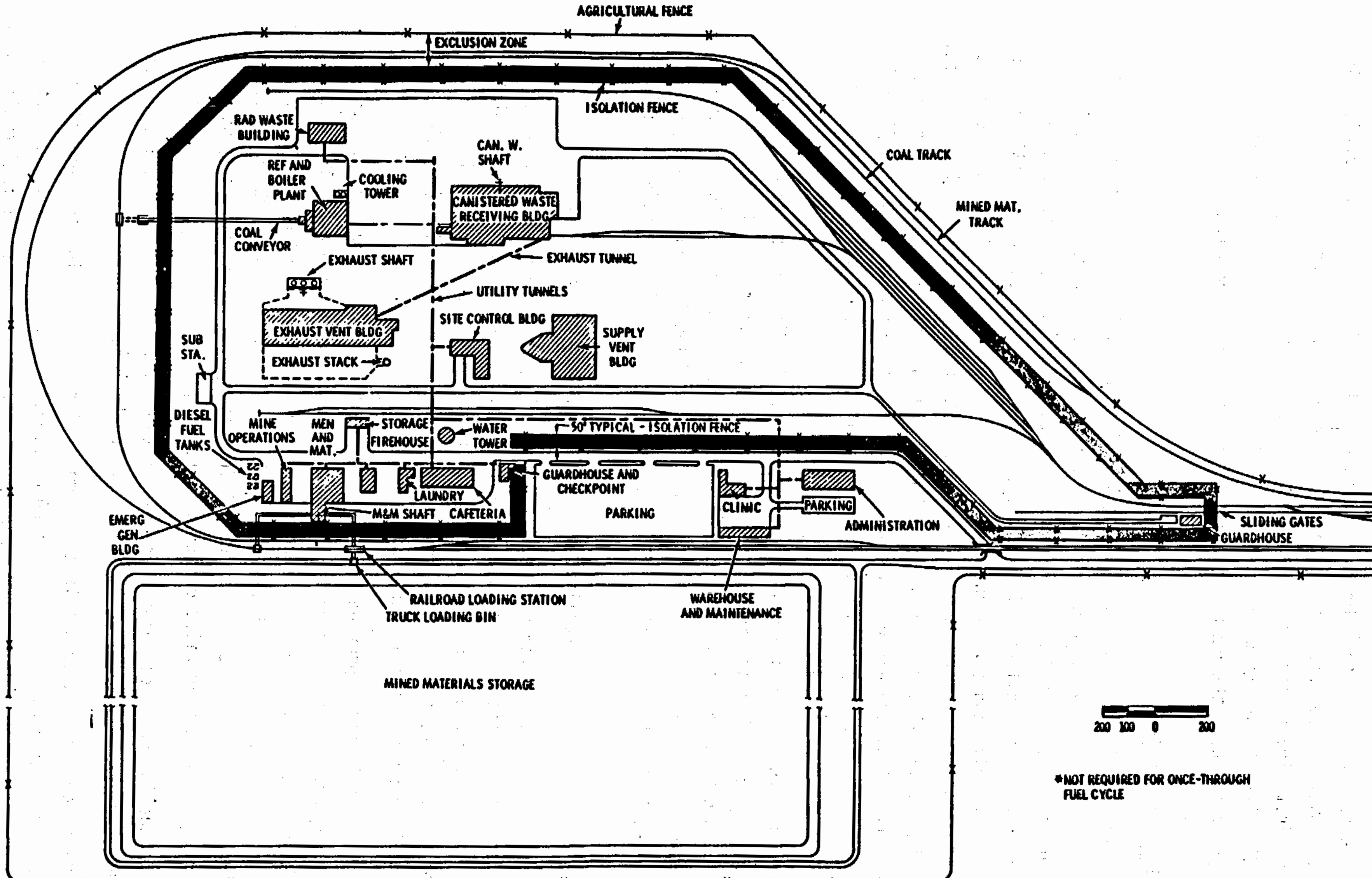
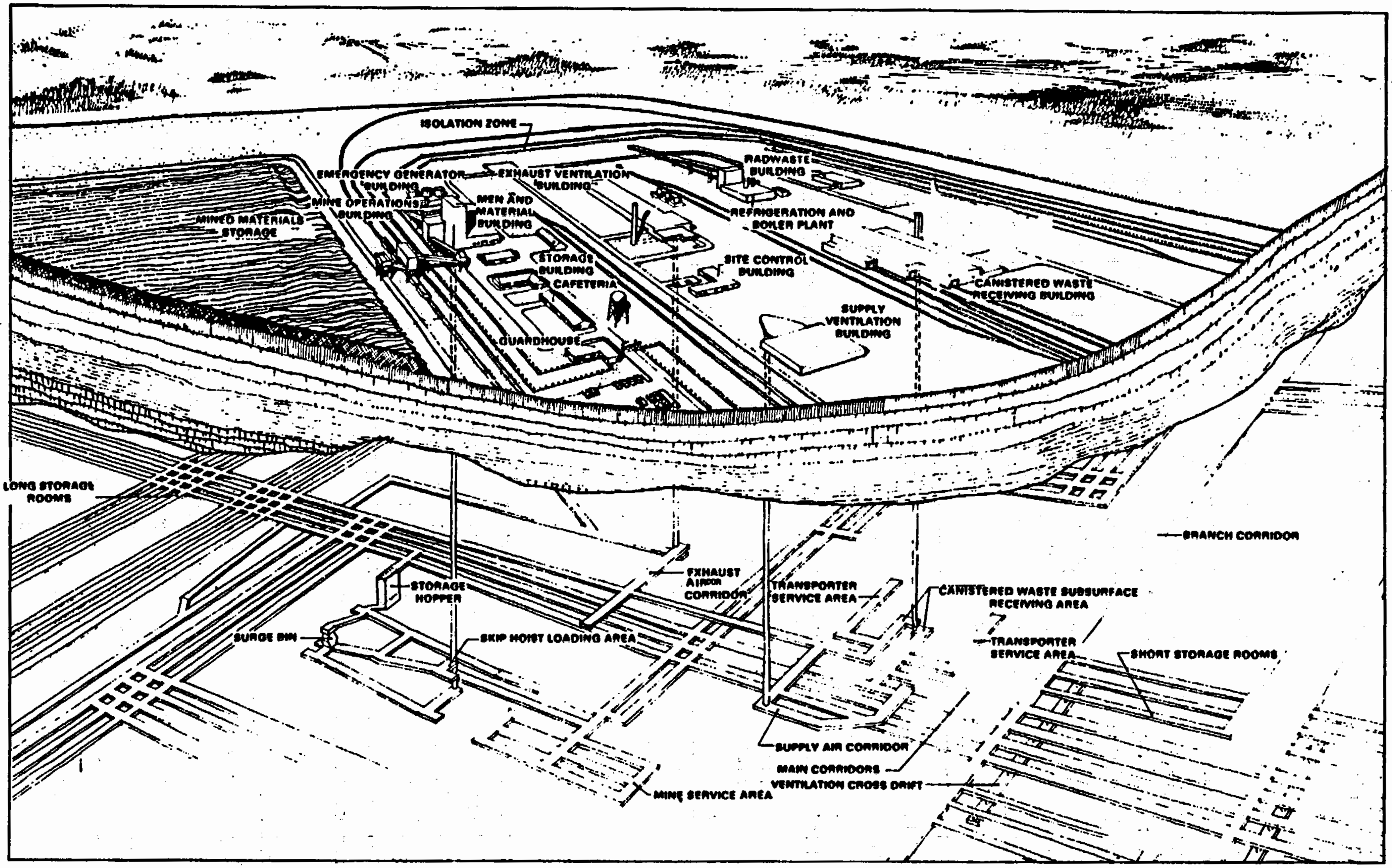


FIGURE 5.3.1. Plot Plan of a Geologic Repository



WASTE ISOLATION FACILITY

FIGURE 5.3.2. Artist's Concept of a Geologic Repository and Its Support Facilities

areas occupy 650 to 730 ha, with the remaining 80 to 160 ha occupied by shafts, general service areas, main corridors and unmined areas within the repository.

5.3.1.3 Construction

In the process of excavating repository subsurface areas, all mined rock is brought to the surface and stored onsite. The storage pile is constructed using standard earth-moving equipment. Standard dust control procedures (water sprays, etc.) are employed during construction at all repositories; salt and shale storage piles are also provided with water run-off control. When retrievable emplacement operations are complete, a portion of the rock will be returned to the mine as backfill. Present plans call for rock not used for backfill to remain piled on the surface. While in the case of a salt repository, excess salt may be disposed of by placing it in an abandoned salt mine or by selling the salt for commercial use, these options depend heavily upon the nature of specific sites. (If mined salt were to be used in commerce, the salt could be moved off site before any radioactive waste arrives onsite. Thus there would be no potential for radioactive contamination of the salt.) Quantities of rock removed and stored are described in Table 5.3.2.

TABLE 5.3.2. Mining and Rock Handling Requirements at the Reference Spent Fuel Repository

	Mined Quantity	Room Backfill	Total Backfill	Permanent Onsite Surface Storage	
	$(MT \times 10^6)$	$(MT \times 10^6)$	$(MT \times 10^6)$	$(MT \times 10^6)$	$m^3 \times 10^6$
Salt	30	14	17	13	6.1
Granite	77	29	38	39	15
Shale	35	15	21	14	5.5
Basalt	90	32	46	44	15

Although a repository in any of the four rock media occupies an overall area of 800 ha, larger amounts of rock are removed in constructing repositories in granite and basalt. This is due in part to larger mining extraction ratios (ratio of mined to intact volume). The increased extraction ratios are possible because of greater rock strength that allows the pillar widths to be decreased, resulting in more emplacement rooms and consequently more waste storage per given repository area.

5.3.1.4 Operations

Spent fuel packaging facilities are here assumed to be incorporated in the repository surface facilities but could be a separate facility nearby. Spent fuel elements arrive at the repository's surface facilities by rail or truck in shipping casks designed for fuel transport. These casks are lifted by crane from the rail cars or trailers to shielded transfer cells for remote removal of the spent fuel assemblies. At this point, the assemblies are examined for external contamination, signs of damage, and compatibility with other acceptance criteria. Acceptable assemblies are encased in helium-filled canisters. The helium atmosphere in the canister provides a means for canister leak testing.

Contaminated assemblies are first cleaned, then sealed in a canister; damaged assemblies are returned to their casks, transferred to the overpack cell, and encased in canisters and appropriately sized overpack canisters. The canisters are then transported to the canistered waste shaft and lowered into the repository. All spent fuel handling is done remotely.

The spent fuel canisters are received at subsurface transfer stations where shielded transporters remotely remove the canisters from the transfer stations for delivery to an emplacement room.

In addition to the thermal restrictions discussed in Section 5.3.1.1, room capacity is limited by the minimum allowable hole spacing of 1.8 m (6 ft) center to center. This is a mechanical limit that prevents weakening of the floor by holes spaced too closely together. The conceptual repositories in salt and shale emplace both PWR and BWR canisters in holes, while repositories in granite and basalt emplace PWR canisters in holes and BWR canisters in trenches. Trenches allow the relatively low heat-generating BWR canisters to be spaced more closely together (trenches are not economical for the higher heat-generating PWR canisters). The trenches run the length of emplacement rooms and contain steel racks to maintain the canisters in an upright position. They are backfilled after emplacement sleeves are installed.

Table 5.3.3 lists the contents of the conceptual spent fuel repositories in salt, granite, shale, and basalt formations at the end of emplacement.

TABLE 5.3.3. Contents of the Conceptual Spent Fuel Repositories When Full

	PWR		BWR		Total
	Canisters	MTHM	Canisters	MTHM	MTHM
Salt	68,200	31,500	104,000	19,600	51,100
Granite	162,700	75,100	246,300	46,500	121,600
Shale	86,300	39,800	131,000	24,700	64,500
Basalt	162,700	75,100	246,300	46,500	121,600

Two separate repository design concepts were also developed for the limited quantities of spent fuel, 10,000 MTHM and 48,000 MTHM, produced in the two cases (Cases 1 and 2 in Section 3.2.2) where the nuclear industry is assumed to be severely constrained. Surface facilities are reduced in size and capacity for these reduced requirements and the mined area is reduced in proportion to the quantity of spent fuel sent to disposal.

5.3.1.5 Retrievability

Actions necessary to remove emplaced wastes from a geologic repository depend on the period of repository operations during which removal takes place. Initially, wastes are emplaced in holes lined with steel sleeves and sealed with removable concrete plugs. The sleeves and plugs ensure the canisters remain accessible and minimize corrosion or other damage. During this period the wastes are considered readily retrievable in that they are removable from the repository at about the same rate and with about the same effort as for emplacement. Beyond this initial period of operation, canisters are emplaced without sleeves and rooms are backfilled. During this later period the wastes are considered to be recoverable at considerably greater effort than emplacement.

For the conceptual repositories, readily retrievable emplacement spans the initial 5 years of operation. Repository excavation is completed during this period, and no wastes are emplaced nonretrievably until after the full extent of the repository has been explored. This provides a period for observation of waste-rock interactions when waste and local rock temperatures reach their maximum. Repository operations would also be evaluated during this period and adjustments made if necessary.

The NRC has recently proposed (Federal Register 1980)[a] that the repository should be designed to allow retrieval of wastes for a period of 50 years after termination of waste emplacement. Whether this proposal might lead to a requirement that the wastes be readily retrievable for this period of time or recoverable has not yet been determined.

Although the specific requirements for 50-year retrievability have not yet been determined, requirements for 25-year retrievability have been estimated and the general nature of requirements for 50-year retrievability can be described. The 25-year retrievability requirements are described in Appendix K. They include use of sleeve-lined holes and concrete plugs and reduced thermal loadings for all of the spent fuel canisters. For 50-year retrievability the thermal loadings would probably have to be further reduced. An alternative approach would be to provide continuing ventilation for heat removal to reduce the rock stresses.

A particular concern for a repository in salt is closure of rooms over long retrievability periods due to accelerated "creep" deformation of the salt caused by the waste's heat. This can be compensated for, at least to some extent, by increasing ceiling heights within the repository (7.6 m height for 25-year retrievability versus 6.7 m in height for 5-year retrievability) but this may be a difficult problem for 50-year retrievability.

After repository performance has been adequately verified (after the initial 5 years of operation for these conceptual repositories, or longer if required), it was assumed that wastes would no longer be emplaced in a readily retrievable manner. For the remainder of repository operations, wastes may be emplaced in holes without steel sleeves. As the wastes are emplaced, the holes are filled with crushed rock or some specially selected backfill material. The backfill material may be an adsortive material selected to increase the probability of long-term waste isolation. After a room is filled with waste, it is backfilled with previously excavated crushed rock or with specially selected backfill material. During this period of repository operations, the wastes are considered to be recoverable from the backfilled rooms. Recovery operations are more difficult and costly than retrieval because of the need to remove room and hole backfill. The nature of these operations increases the possibility of waste canisters being damaged before or during recovery operations but conventional techniques should be adequate. It is possible that this condition

(a) Federal Register, Vol. 4, N.94, May 13, 1980, page 31400.

might be considered adequate to meet the intent of the requirements proposed by the NRC. Additional details of retrieval and recovery operations are provided in Appendix K.

5.3.1.6 Decommissioning

As mentioned in Section 5.3.1.5, after the readily retrievable period, rooms that have been filled to capacity with spent fuel are backfilled. The technique selected for the conceptual repository is to fill the rooms with previously excavated crushed rock or with specially selected backfill material. Standard earth-moving equipment will be used to do this. This technique was selected as the most economical, and it reduces the amount of mined rock stored on the surface. With this technique, the rooms are backfilled to within 0.6 m of the ceiling with crushed rock at approximately 60% of its original density. Other backfill materials and methods of emplacement are discussed in Koplick et al. (1979).

After all rooms have been filled with spent fuel and are backfilled, the remainder of the repository underground areas are decommissioned. All corridors and underground areas are backfilled in the same manner as emplacement rooms. After this is completed, the repository shafts are decommissioned by filling to the surface and sealing. Combinations of crushed rock, clay, and concrete may be used for this purpose. Because the procedures to be used are highly site and media specific, they are not described in this generic Statement (see Koplick et al. (1979)).

Repository decommissioning is complete when the surface facilities are decontaminated, perhaps dismantled, and the repository location is monumented.

5.3.2 Reprocessing Fuel Cycle Repository

A geologic repository operating for disposal of fuel reprocessing wastes in the reprocessing fuel cycle would be required to receive high-level waste (HLW) and various remotely handled TRU (RH-TRU) and contact-handled TRU (CH-TRU) wastes. The characteristics of these wastes from reprocessing commercial fuel are described in Section 4.3. Defense program wastes could be accommodated in geologic repositories in a manner similar to that described here for these commercial fuel cycle reprocessing wastes. Characteristics and quantities of these wastes are described in Appendix I. While these latter wastes differ from those from LWR fuel reprocessing, the differences (mainly older and cooler, smaller quantities of high-atomic-number actinides and different chemical form) produce wastes with lower radiation intensities and lower heat output. Thus, repository placement criteria would be less stringent for defense wastes than those for commercial wastes and they could therefore be accommodated in the same repositories.

5.3.2.1 Design Bases

As described in Section 5.3.1.1 for the once-through fuel cycle repository, waste emplacement is subject to thermal loading criteria for a given type of waste and rock. The limits listed in Table 5.3.4 for the reprocessing fuel cycle repository are two-thirds of the calculated permissible criteria described in Appendix K.

In the case of reprocessing cycle high-level wastes there is a thermal limit for individual canisters in addition to the repository area thermal limits. These limits, which are derived from maximum temperatures, are identified in Table 5.3.5.

TABLE 5.3.4. Conceptual Repository Design Thermal Limits for Reprocessing Cycle Wastes

Medium	kW/ha[(a)]	kW/acre[(a)]
Salt[(b)]	250	100
Granite	320	130
Shale	200	80
Basalt	320	130

(a) Area occupied by the emplacement rooms and their associated pillars only.
(b) The placement of HLW in salt is not limited by long-term surface uplift as was the case for spent fuel in salt. Because the concentration of plutonium and its long-term heat contribution is much less in HLW, surface uplift is reduced and room and pillar integrity is the dominant concern. The integrity of rooms and pillars is dependent upon room and pillar area thermal density as listed in this table

TABLE 5.3.5. Conceptual Repository Thermal Limits for Individual HLW Waste Canisters

Medium	Maximum kW per Canister
Salt	3.2
Granite	1.7
Shale	1.2
Basalt	1.3

The conceptual repositories are designed to receive and emplace 6.5-year-old (time since reactor discharge) HLW. However, as was the case with spent fuel (Section 5.3.1.1), much of the HLW as it arrives at the repository will be older and cooler than 6.5 years. Because of this, estimates of waste emplacement for the reprocessing waste repositories are conservative because the repository could hold more waste if designed for the older and lower heat-generating rate wastes. As in the case of the spent fuel criteria, the criteria in Table 5.3.4 were developed for 10-year-old waste. Using these criteria for 6.5-year-old waste provides additional conservatism here also. However, the effect on capacity is smaller here because a substantial portion of the repository area is required for TRU wastes whose placement is not affected by the thermal criteria because they generate so little heat.

Design and construction of the conceptual fuel reprocessing waste repositories are assumed to proceed in the same manner as described for the once-through fuel cycle in Section 5.3.1.1. The overall repository area is approximately 800 ha in all cases. Construction is completed during the first five years of repository operations while all wastes are emplaced retrievably.

5.3.2.2 Facility Description

The conceptual repositories consist of surface and subsurface facilities. The surface facilities provide for waste receiving and handling, mining and general operations support. The subsurface facilities provide for waste handling and storage and mined rock removal. The surface facilities and the mined rock storage pile constitute the visible evidence of the repository and occupy an area of about 180 ha at the salt and shale repositories and 220 ha at the granite and basalt repositories. These quantities vary slightly from the spent fuel case because of different repository configurations and mining extraction ratios.

Additional details of repository surface facilities are given in DOE/ET-0028.

The conceptual geologic repositories for the fuel reprocessing wastes require the shafts described in Section 5.3.2.2 for the once-through fuel cycle repositories and an additional CH-TRU waste shaft to transfer the waste from the CH-TRU waste building to the subsurface emplacement area.

The repository underground layout is a conventional room and pillar arrangement that serves the need for repository ventilation, opening stability, thermal effects and efficient use of excavated space. Of the 800-ha total area, actual waste emplacement areas occupy 650 to 730 ha, with the remaining 80 to 160 ha occupied by shafts, general service areas, main corridors and unmined areas within the repository.

5.3.2.3 Construction

As for the once-through fuel cycle repository, all mined rock is brought to the surface during repository excavation. Mining and rock handling requirements for the conceptual repositories in the four media are compared in Table 5.3.6. The larger amounts of mined rock in granite and basalt are the result of increased mining extraction ratios in these geologies. As in the once-through cycle there is the possibility of selling the excess salt for commercial use in the case of a salt formation repository.

TABLE 5.3.6. Mining and Rock Handling Requirements at the Reference Reprocessing Waste Repository

	Mined Quantity	Room Backfill	Total Backfill	Permanent Surface Surface Storage	
	($MT \times 10^6$)	($MT \times 10^6$)	($MT \times 10^6$)	($MT \times 10^6$)	($m^3 \times 10^6$)
Salt	35	15	20	15	7.1
Granite	53	17	24	29	11
Shale	30	12	17	13	5.1
Basalt	59	17	27	32	11

5.3.2.4 Operations

Canisters of HLW, and RH-TRU wastes are received and handled at the repository in a similar manner to that previously described for spent fuel in the once-through fuel cycle repository. Canisters found to be damaged or leaking are taken to an overpack cell and sealed in an appropriately sized overpack canister. RH-TRU waste in 55-gal drums is shipped to the repository by truck, arriving in shielded Type B overpacks (see Section 4.5.3.2 for Type B overpack definition). The overpacks are lifted by crane from the truck bed to shielded transfer cells for remote removal of the drums. The drums are placed three each into steel drum-pack canisters which are sealed with a welded lid. The drum-pack is transported to the canistered waste shaft and lowered into the repository.

CH-TRU waste arrives at the repository on pallets of twelve 55-gallon drums stacked two by three by two drums high or in steel boxes measuring 1.2 x 1.8 x 1.8 m (4 x 6 x 6 ft), roughly equivalent in size to the pallet of drums. The CH-TRU is shipped by truck in special cargo carriers (see Section 4.5) loaded with three pallets or boxes of waste. The pallets and boxes are unloaded from the cargo carrier using shielded forklifts, inspected for damage and repaired if necessary, transported to the CH-TRU waste shaft and lowered into the repository.

Wastes are received at subsurface transfer stations that form integral structures with the shafts. Shielded transporters remotely remove the containers from the transfer stations for delivery to an emplacement area.

At the conceptual repositories in salt and shale formations, HLW canisters are lowered into vertical holes in the emplacement rooms in accordance with the same minimum hole spacing (1.8 m) described for spent fuel canisters in the once-through fuel cycle repositories and with an allowable thermal density calculated specifically for the HLW's characteristics. In these formations, RH-TRU waste is also emplaced in drilled holes; however the minimum hole spacing is increased to 2.3 m as a result of the larger-hole diameters necessary for the 0.76-m-diameter canisters.

The conceptual repositories in granite and basalt formations emplace HLW in vertical holes as described for the salt and shale repositories. However, RH-TRU canisters are lowered into trenches running the length of the rooms. The canisters are held upright in a single row by storage racks that allow a minimum spacing of 1 m center-to-center.

Shielded forklifts stack the CH-TRU waste pallets and boxes two high along the walls of CH-TRU waste emplacement rooms.

Table 5.3.7 lists the contents based on the example treatment processes described in Section 4.3 of conceptual repositories located in salt, granite, shale, and basalt formations at the end of operations. Because of the differences in thermal criteria the capacities of different rock media vary. For the conceptual repositories illustrated here, the relative quantities of high-level waste and TRU wastes are different on an MTHM-equivalent basis. This is because the five-year cooling hold up for the HLW resulted in a disproportionately larger quantity of TRU waste being emplaced. Subsequent repositories would fill up with more nearly equivalent amounts of HLW and TRU wastes. The capacities when equivalent quantities of HLW and TRU wastes are emplaced are also shown.

TABLE 5.3.7. Contents of the Conceptual Reprocessing Waste Repositories When Full

Waste	Salt		Granite		Shale		Basalt	
	Containers	Equivalent MTHM[(a)]	Containers	Equivalent MTHM[(a)]	Containers	Equivalent MTHM[(a)]	Containers	Equivalent MTHM[(a)]
HLW Canisters	25,800	62,200	48,000	69,000	36,000	30,500	63,000	56,000
RH-TRU Canisters	26,900		29,100		15,100		24,700	
RH-TRU Drums	399,000	99,700	431,000	108,500	224,000	56,000	367,000	91,500
CH-TRU Boxes	4,150		4,500		2,290		3,810	
CH-TRU Drums	264,000		286,000		144,000		242,000	
Capacity if Equivalent Quantities of HLW and TRU Wastes are Emplaced		71,200		78,600		41,100		73,800

(a) For the conceptual repositories the relative quantities of HLW and TRU wastes are different because the HLW is held up for a 5-year cooling period allowing a disproportionate quantity of TRU waste emplacement. The third number shows the capacity when both waste types are emplaced at the same equivalent rates.

5.3.2.5 Retrievability

These conceptual repositories are operated with the same initial period of retrievability described for the once-through fuel cycle repositories. Steel sleeves and concrete plugs are used as described for the spent fuel to protect the emplaced HLW and RH-TRU waste canisters during the retrievable period. CH-TRU waste does not require this additional protection because it is stacked compactly in the emplacement room rather than being placed into drilled holes.

5.3.2.6 Decommissioning

Reprocessing fuel cycle repositories are decommissioned in the same manner described in Section 5.3.1.6 for the once-through fuel cycle repositories.

5.3.3 Effect of Waste Age on Repository Capacity

As spent fuel or HLW ages, the intensity of emitted radiation and heat declines and the quantity of these materials that can be emplaced in a given repository area increases somewhat. Although the thermal loading criteria for a given temperature limit decreases with waste age, heat emissions from the waste decrease even faster so that the overall result is an increase in repository capacity with increasing waste age.

The thermal loading limit for 10 year old waste is smaller than the limit for younger waste (See Appendix K for details). For a fixed initial repository thermal loading, the quantity of waste is smaller and less heat will be emitted over the long term with 6.5 year old waste than with 10 year old waste. The capacities for the conceptual repositories described in the previous sections were based on 6.5-year-old spent fuel and high-level waste, conservatively employing the thermal loading criteria for 10-year-old waste. These conceptual designs were used as a conservative basis to develop environmental impacts, resource requirements and costs for individual repositories. However, in the system simulation calculations in Chapter 7, where spent fuel and HLW ages range up to more than 50 years for some of the delayed repository cases, the repository requirements are based on estimated thermal limits that vary with waste age. The limits used are two-thirds of the estimated maximum allowable loadings.

The calculated relationship between repository capacity and waste age is shown in Figure 5.3.3 for the once-through cycle and in Figure 5.3.4 for the reprocessing cycle. The capacity of a salt repository for spent fuel is indicated to be substantially less than for reprocessing wastes and increases only about 10% over the age range shown here. (Spent fuel emplacement in salt is limited by surface uplift from the long-term heat generation from the contained plutonium. This is not a problem with the other media.) Increases in capacity for the other media range from 30% for spent fuel in shale to 100% for reprocessing wastes in granite. Repository capacities for spent fuel are more than reprocessing waste capacities in granite, basalt, and shale. This is primarily because of the repository area required for TRU wastes, which ranges from 30% for 5-year-old HLW to as much as 70% in granite and basalt for 50-year-old HLW. Design optimization and/or treatments that reduce TRU waste volumes might mitigate this effect.

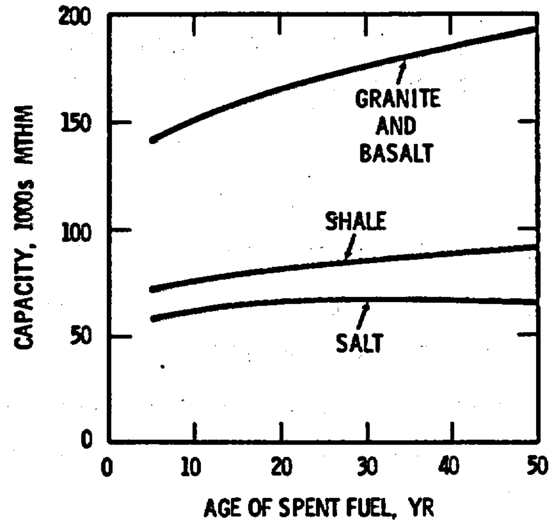


FIGURE 5.3.3. Effect of Spent Fuel Age on Once-Through Cycle Repository Capacities

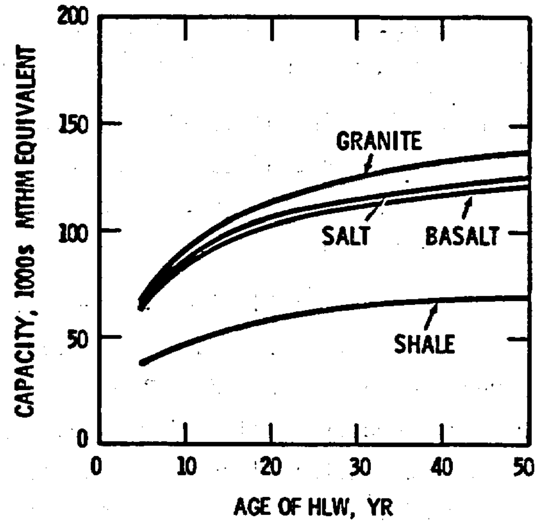


FIGURE 5.3.4. Effect of HLW Age on Reprocessing Cycle Repository Capacities

Further details regarding the basis and derivation of these repository capacities are provided in Appendix K.

5.3.4 Regional Repository Concept[a]

To the extent permitted by availability of suitable geologic sites, two or more repositories could be located to provide disposal services on a regional basis. A regional siting concept for geologic repositories was proposed by the Interagency Review Group (IRG) on Nuclear Waste Management (IRG 1979). In its Report to the President, the IRG recommended construction of several repositories sited on a regional basis insofar as technical consid-

(a) Section 2.3 describes the present National Site Characterization and Selection Plan. Section 5.2 and Appendix B.7 discuss the technical considerations of repository site selection.

erations permit, as opposed to a single national repository. This strategy would integrate societal and political concerns as well as technical considerations. Possible advantages of the regional concept include:

- More equitable distribution of waste management costs;
- Enhanced ability to gain public and political acceptance through cooperative participation with state and local officials and groups;
- Experience with various environments and emplacement geologic media sooner than previously planned, especially with near simultaneous development of several repositories; and
- Reduction of transportation requirements and attendant risks.

Definition of regions for nuclear waste isolation can be influenced by a number of technical, societal, and political factors. The major technical factor is the geographic distribution of acceptable geologies, but a number of other factors must also be considered.

An obvious regional division of the U.S. is one based upon individual states or combinations of states. The predominant factors that affect regional boundaries derived from the boundaries of states are the historical, social, geographical, and political factors that have existed to define the states themselves.

Regions established strictly on existing political or commercial factors could yield a wide region-to-region variation in the quantities of waste generated. Thus, there is some incentive to develop a regional structure that is based on reasonably uniform waste generation. Locations of nuclear generating capacity or electrical usage may provide an equitable basis for regional structures. Extensive electrical grid interconnections may extend the use of nuclear generated power far beyond plant locations and should be considered.

Although multiple sites themselves (except to the extent provided by different geologies) provide no guarantee against errors in disposal technology or repository design, they do help minimize the consequences of errors if the resulting failures are random and widely spaced in location and time (i.e., well after the repositories have been sealed). The potential for reduced consequences lies in the possibility of some repositories remaining unaffected, and the use of knowledge gained from the first incident to prevent subsequent incidents at other locations.

While at the present time the Department of Energy is not able to propose a specific regional siting program, regional siting is presently considered, among other factors, in the site-selection process. The Department is continuing to study the regional siting concept and should a regional siting plan be adopted, the data from the first repository could be incorporated in such a plan.

REFERENCES FOR 5.3

Interagency Review Group. 1979. Report to the President by the Interagency Review Group on Nuclear Waste Management, TID-29442. U.S. Department of Energy, Washington, D.C.

Koplick, C. M. et al. 1979. Information Base for Waste Repository Design, Vol. 1 and 5, NUREG/CR-0495 (7 volumes). Analytic Sciences Corp. for the U.S. Nuclear Regulatory Commission, Reading, Massachusetts.

U.S. Department of Energy. 1979. Technology for Commercial Radioactive Waste Management, Vol. 4, DOE/ET-0028 (5 volumes). Washington, D.C.

5.4 ENVIRONMENTAL IMPACTS RELATED TO REPOSITORY CONSTRUCTION AND OPERATION

Environmental impacts related to repository construction are those estimated for construction of surface facilities and mining of the entire repository, whereas those for operation are associated with waste emplacement, backfilling and decommissioning of surface facilities. Additional details are presented in DOE/ET-0029.

5.4.1 Resource Commitments

Land use commitments for single conceptual repositories in the four geologic media are summarized in Table 5.4.1 for both spent fuel and reprocessing wastes. Other resource commitments are tabulated in Table 5.4.2 for spent fuel repositories and in Table 5.4.3 for reprocessing waste repositories. The same size (areal extent) of repository (800 ha) is postulated for each rock type; however, thermal criteria (heat loading of rock) allow spent fuel containers to be stored closer together in granite and basalt than in salt and shale, thus greater quantities of high-level waste can be stored in granite and basalt repositories for a given area than in salt and shale repositories.[a]

TABLE 5.4.1. Land Use Commitments For Construction of 800-ha Single Geologic Repositories

Land Use	Salt & Shale	Granite & Basalt
Surface facilites, ha		
Spent fuel repository	180	280
Reprocessing waste repository	180	220
Access roads and railroads, ha	8	8
Mineral and surface rights, ha (fenced restricted area)	800	800
Additional land on which only subsurface activities will be restricted, ha	3,200	3,200

Land use conflicts will be highly site specific; however, most restrictions on surface use of land need not continue after repository closure. Thus, most uses of the land could resume after decommissioning of the surface facilities.

Water used during construction of a repository will range from about 1×10^5 to 5×10^5 m^3 (depending on geologic medium) over the 7-yr construction period. As long as water can be supplied from rivers such as the R River in the midwest reference environment (Appendix F), water use will represent a small fraction (0.001) of the average river flow

(a) Note, however, that waste emplacement has not been optimized in an engineering sense for this generic Statement.

TABLE 5.4.2. Resource Commitments Necessary for Construction of a Spent Fuel Repository in Salt, Granite, Shale, and Basalt

Resource	Salt (51,000 MTHM)	Granite (122,000 MTHM)	Shale (64,000 MTHM)	Basalt (122,000 MTHM)
Water Use, m^3	240,000	710,000	360,000	610,000
Materials				
Concrete, m^3	100,000	300,000	150,000	250,000
Steel, MT	16,000	48,000	24,000	40,000
Copper, MT	220	660	330	560
Zinc, MT	55	160	80	140
Aluminum, MT	41	120	64	110
Lumber, m^3	2,300	6,900	3,000	5,900
Energy Rescources				
Propane, m^3	2,200	6,400	3,200	5,400
Diesel fuel, m^3	22,000	64,000	32,000	54,000
Gasoline, m^3	16,000	47,000	21,000	40,000
Electricity				
Peak demand, kW	3,400	11,000	5,100	8,800
Total consumption, kWh	14,000,000	43,000,000	21,000,000	36,000,000
Manpower, man-yr	10,000	30,000	14,000	37,000

TABLE 5.4.3. Resource Commitments Necessary for Construction of a Fuel Reprocessing Waste Repository in Salt, Granite, Shale, and Basalt[(a)]

Resource	Salt (62,000 MTHM HLW)	Granite (69,000 MTHM HLW)	Shale (30,000 MTHM HLW)	Basalt (56,000 MTHM HLW)
Water use, m^3	270,000	510,000	290,000	450,000
Materials				
Concrete, m^3	110,000	210,000	120,000	190,000
Steel, MT	18,000	33,000	19,000	30,000
Copper, MT	240	470	260	420
Zinc, MT	62	120	67	110
Aluminum, MT	46	90	50	77
Lumber, m^3	2,600	4,900	2,800	4,400
Energy resources				
Propane, m^3	2,400	4,500	2,600	4,000
Diesel fuel, m^3	24,000	45,000	26,000	40,000
Gasoline, m^3	18,000	33,000	19,000	30,000
Electricity				
Peak demand, kW	3,900	7,300	4,100	6,600
Total Consumption, kWh	16,000,000	30,000,000	17,000,000	27,000,000
Manpower, man-yr	11,000	22,000	13,000	26,000

(a) Only HLW are indicated in this and subsequent tables referring to reprocessing wastes sent to repositories. In addition to HLW, about 100,000 MTHM equivalent of TRU wastes are placed in the "first" salt repository and about 110,000, 56,000 and 92,000 MTHM equivalent in "first" repositories in other media, respectively. Subsequent repositories would undoubtedly receive a different mix of HLW and TRU wastes.

and no significant impacts are expected from its withdrawal. If a repository was to be built in an arid region, water might need to be transported to the site from areas of abundant supply.

5.4.2 Nonradiological Effluents

Nonradiological effluents from repository construction include dust and pollutants generated from machinery operation during surface facility construction and mining operations. Burning the quantities of fossil fuels listed in Tables 5.4.2 and 5.4.3 results in air pollutant emissions, but concentrations in air at the fenceline are not expected to result in any air quality degradation outside applicable limits (40 CFR 50). Estimates of pollutant totals released to the atmosphere from operating equipment during construction are given in Table 5.4.4. These quantities are developed from the total quantities of fuel burned and emission factors for a given effluent (URS 1977).

TABLE 5.4.4. Quantities of Effluents Released to the Atmosphere During Construction of a Geologic Repository

Pollutant, MT	Salt	Granite	Shale	Basalt
	for Spent Fuel			
	(51,000 MTHM)	(122,000 MTHM)	(64,000 MTHM)	(122,000 MTHM)
CO	7,900	23,000	10,000	20,000
Hydrocarbons	360	1,100	480	890
NO_x	1,500	4,500	2,200	3,800
SO_x	92	270	130	230
Particulates	94	270	130	230
	for Reprocessing Wastes			
	(62,000 MTHM)	(69,000 MTHM)	(30,000 MTHM)	(56,000 MTHM)
CO	8,800	16,000	9,300	15,000
Hydrocarbons	400	740	420	660
NO_x	1,700	3,100	1,800	2,800
SO_x	100	190	110	170
Particulates	100	190	110	170

Fmissions from oil burning space heaters in a town of 30,000 population (about 8,000 heaters) were estimated for a 20-yr period (the approximate time surface facilities at a repository are operating) in an effort to provide some perspective for effluents released during construction of a repository. The calculated emissions were:

CO, MT	220
Hydrocarbons, MT	120
NO_x, MT	540
Particulates, MT	6,000
SO_x, MT	460

Dust from mining and rock transport within the mine is removed by filters in the mine ventilation system. However, dust generated from surface operations and rock transport to storage will result in above-ground dust. Potential dust emissions were determined using emission factors estimated by Cowherd et al. (1974). These factors were measured for rock aggregate storage piles (but not for salt) under dry and windy conditions when the dust generating potential was near maximum. Table 5.4.5 presents dust emissions for the various host rock types for both the reference environment (moist regions) and arid regions.

TABLE 5.4.5. Maximum Dust Emissions From Surface Handling of Mined Material, MT/d[a]

Climate	Spent Fuel Repository: Salt (51,000 MTHM)	Granite (122,000 MTHM)	Shale (64,000 MTHM)	Basalt (122,000 MTHM)
Reference	3.1	7.9	3.7	9.3
Arid	44	110	51	130
	Reprocessing Waste Repository			
	(62,000 MTHM)	(69,000 MTHM)	(30,000 MTHM)	(56,000 MTHM)
Reference	3.6	5.6	3.1	6.1
Arid	49	79	44	86

(a) Assuming no control techniques are applied.

The maximum and average concentrations of dust at the repository fenceline (1.6 km from repository center) were calculated using the average annual dispersion factors (X/Q') presented for the reference environment. Table 5.4.6 presents these concentrations for the four geologic media.

The existing primary Federal air quality standard for suspended particulate matter computed as an annual geometric mean is 75 g/m^3. Thus, for both the reference site and any proposed arid site, appropriate control techniques will be necessary to assure this limit is not exceeded during surface handling of mined material.

To give perspective to the salt concentrations at the repository fenceline, as given in Table 5.4.6, note that nearshore salt concentrations on the eastern seaboard average about 140 $\mu g/m^3$ at 0.5 km inland and about one-tenth of that 1 km inland. During persistently high onshore winds, the concentration may be on the order of 380 $\mu g/m^3$ at 0.5 km and 60 $\mu g/m^3$ at 1 km (CONF 740302 1974, pp 353-369).

Table 5.4.7 presents estimates of dust deposition rates from surface handling of mined material. Maximum deposition of dust would occur at a distance of 0.4 km from surface handling operations. At the repository fenceline (1.6 km from the handling operations) deposition is approximately a factor of 10 less. These depositions are based on the "worst case," which would consider the maximum removal rate for a year's period. Impacts of these depositions were they to occur are discussed later in the section on evaluating ecological effects of repository construction.

TABLE 5.4.6. Particulate Concentrations at Repository Fenceline, $\mu g/m^3$(a)

Spent Fuel Repositories: Repository Medium	Maximum	Average	Reprocessing Waste Repositories: Repository Medium	Maximum	Average
Salt			Salt		
• Reference environment	110	66	• Reference environment	130	71
• Arid environment	1400	790	• Arid environment	1600	930
Granite			Granite		
• Reference	290	170	• Reference	200	120
• Arid	3500	2100	• Arid	2400	1400
Shale			Shale		
• Reference	130	79	• Reference	110	66
• Arid	1600	930	• Arid	1400	790
Basalt			Basalt		
• Reference	330	190	• Reference	210	130
• Arid	4100	2400	• Arid	2600	1600

(a) Assuming no control techniques are applied.

TABLE 5.4.7. Dust Depositions from Surface Handling of Mined Material, gm/m^2-yr(a)

	Spent Fuel Repository		Reprocessing Waste Repository	
	Distances from Handling Operations			
	0.4 km	1.6 km	0.4 km	1.6 km
Salt				
• Reference environment	70	8.4	90	11
• Arid environment	870	84	1100	110
Granite				
• Reference	180	22	140	17
• Arid	2200	220	1700	170
Shale				
• Reference	82	9.8	79	9.7
• Arid	1000	98	970	97
Basalt				
• Reference	210	25	160	19
• Arid	2600	250	1900	190

(a) Assuming no control techniques are applied.

The main concern related to surface stockpiles would be the need to protect the ground and surface waters from being contaminated with stockpile runoff, particularly in the case of salt. For repositories in salt, one plan calls for an impermeable lining of hypalon covered by 2 ft of montmorillonite-type clay to be placed over the entire stockpile area after grading and before stockpiling begins. The hypalon and clay function as a groundwater protection barrier. Construction of a trench with the same type of protection around the stockpile could collect runoff water and transport it for any required treatment. If the mine is located in an area with an arid climate, an evaporation pond may provide the required treatment. If an evaporation pond is not practical, the runoff water may be drained into a sump and pumped to a water treatment plant where dissolved salt or other solids could be removed.

Several methods for disposing of salt in excess of needs for backfilling have been investigated (D'Applonia 1976). These included disposal at sea, backfilling abandoned mines, and use in the salt trade. Salt stockpiles crust quickly and industry does not spread asphalt or chemicals on top of stockpiles to prevent loss of salt through erosion. However, covering the piles with asphalt or rock and earth may be an appropriate means of assuring dust control in the long term. Several methods appear to control or satisfactorily reduce movement of salt by wind and water. The DOE recognizes the potential for contamination of land by salt and, if a repository is located in salt, is committed to its proper control or suitable disposal.

Shale could conceivably contain amounts of soluble minerals that would be detrimental to the environment. Precipitation could leach these minerals and pollute surface and ground waters. Moreover, in a cold climate, freezing of the wet rock might result in fragmentation and liberation of particulates, resulting in particulate pollution of the streams. The shale stockpile area could be covered with a blanket of montmorillonite clay and sloped toward a collecting ditch. The surface water would then drain into a settling pond to collect silt and sands. From the pond it would be pumped to a water treatment plant where minerals in solution would be removed before release until surface facilities are decommissioned. (At present no provision is made for water treatment after the surface facilities have been decommissioned.)

Granite and basalt generally do not contain noxious soluble substances. Therefore, the stockpile area would not need special treatment and surface water would not have to be treated.

Sanitary waste will be collected in a sewer system that is connected to a local sewer trunk, if available, or given secondary treatment at the repository and disposed of in accordance with local and Federal regulations. Storm drains will be separate from the sanitary sewer system and will lead to a storm drainage pond in the general yard area.

Although dust and nonradiological pollutants generated during construction have a recognized potential for temporary adverse effects, with proper control measures, no long-term effects are expected to result.

5.4.3 Radiological Effects

The release to the atmosphere of naturally occurring radon and its decay products will increase during mining of the repositories. Estimated quantities of these radionuclides likely to be released annually to the biosphere for the various geologic media are listed in Tables 5.4.8 and 5.4.9.

TABLE 5.4.8. Annual Releases of Naturally Occurring Radionuclides to Air for Construction of Geologic Repository for Spent Fuel, Ci

	Geologic Media			
Nuclide	Salt (51,000 MTHM)	Granite (122,000 MTHM)	Shale (64,000 MTHM)	Basalt (122,000 MTHM)
^{220}Rn	9.3×10^{-4}	2.0×10^{1}	6.1	3.1
^{222}Rn	1.3×10^{-3}	1.9×10^{1}	7.0	2.7
^{210}Pb	1.1×10^{-7}	1.6×10^{-3}	5.9×10^{-4}	2.3×10^{-4}
^{212}Pb	1.4×10^{-6}	3.0×10^{-2}	9.2×10^{-3}	4.7×10^{-3}
^{214}Pb	1.3×10^{-3}	1.9×10^{1}	7.0	2.7
^{210}Bi	1.3×10^{-3}	1.9×10^{1}	7.0	2.7

TABLE 5.4.9. Annual Releases of Naturally Occurring Radionuclides to Air for Construction of Geologic Repository for Fuel Reprocessing Waste, Ci

	Geologic Media			
Nuclide	Salt (62,000 MTHM)	Granite (69,000 MTHM)	Shale (30,000 MTHM)	Basalt (56,000 MTHM)
^{220}Rn	1.1×10^{-3}	1.4×10^{1}	5.1	2.0
^{222}Rn	1.6×10^{-3}	1.3×10^{1}	6.0	1.7
^{210}Pb	1.3×10^{-7}	1.1×10^{-3}	2.5×10^{-4}	1.4×10^{-4}
^{212}Pb	1.7×10^{-6}	2.1×10^{-2}	7.7×10^{-3}	3.0×10^{-3}
^{214}Pb	1.6×10^{-3}	1.3×10^{1}	6.0	1.7
^{210}Bi	1.6×10^{-3}	1.3×10^{1}	6.0	1.7

A summary of 70-yr whole-body doses to the construction work force and to the regional population from the releases of "enhanced" quantities of naturally occurring radionuclides is given in Table 5.4.10.

The 70-yr dose from undisturbed naturally occurring radionuclides is about 7 rem/person. The 70-yr dose to the regional population is about 14,000,000 man-rem from undisturbed naturally occurring sources.

In this report, 100 to 800 health effects are postulated to result in the exposed population per million man-rem. Based on the calculated doses to the regional population, no health effects are expected to result from construction of a geologic repository for spent fuel or for reprocessing wastes.

TABLE 5.4.10. Summary of 70-Yr Whole-Body Dose Commitments from Naturally Occurring Radioactive Sources During Mining Operations at a Repository, man-rem

	Spent Fuel Repositories			
	Salt	Granite	Basalt	Shale
Repository				
Work force (7 yr in the repository mine)	0.18	5000	6200	1900
Population (within 80 km)	0.007	100	15	38

5.4.4 Evaluation of Ecological Impacts Related to Repositories[a]

Construction of surface facilities at repositories will involve the removal of vegetation and displacement of birds and small mammals from the site areas. Weedy species of plants would invade cleared areas unless revegetation practices are applied. Localized dust problems would occur until vegetation cover is re-established.

Soil erosion control measures will be needed to prevent surface runoff from adding suspended solids to nearby land and surface waters. If only reasonably good practices were used, effects from construction of the surface facilities on aquatic biota should be negligible.

5.4.4.1 Ecological Effects Related to Repositories in Salt

The major ecological impact would be from fugitive dust depositions which might occur from surface handling operations of mined material. Of most concern are the estimated salt depositions at the repository fenceline of 8.4 and 84 g/m^2-yr for the reference and arid environment, respectively. These depositions were calculated from the case where 3.0×10^7 MT of salt was mined with 1.3×10^7 MT remaining on the surface for final disposal.

Adverse biotic effects on vegetation would depend upon many factors, including rate of uptake, short- and long-term sensitivity of species to effluent concentrations, period of exposure, the physiological condition of the vegetation during the time exposure and buildup of salt over time. Impingement upon vegetation with subsequent foliar absorption appears to be the most hazardous mode of entry. Uptake of salt solutions by foliage is a rapid and relatively efficient process (Bukocac and Wittier 1957). Crops particularly sensitive to salt effects are alfalfa, oats, clover, wheat, Indian rye grass, and ponderosa pine. These plants are seriously damaged during germination and young-leaf stage development. Ornamental vegetation types that are susceptible to salt concentrations are dogwood, red-maple, Virginia creeper and wild black cherry. Visual symptoms of toxicity are foliar necrosis, short-time dieback and "molded" growth habits. Beans are particularly sensitive showing wilting of areas on primary leaves followed by necrosis of previously wilted areas and

(a) In the following discussion of ecological impacts it is assumed that no precautions are taken. Impacts presented can be reduced to insignificant levels through application of available engineering techniques. DOE is committed to discovery and resolution of any potentially significant specific ecological effects.

chlorosis of young trifoliate leaves. Effects on vegetation will depend on air concentration and time of exposure as well as humidity. Generally, an air concentration above 10 $\mu g/m^3$ will alter distribution and growth of plants (Bernstein and Hayward 1958). Because fenceline ground level concentrations for salt dust released from surface storage and handling operations will exceed this level, a significant affect would be expected. The deposition rates are in the range of 40 to 95 $gm/m^2/yr$ for observable leaf-burn on such plants as beans. Based on the assumptions made for determining salt depositions, mitigating procedures would be needed to reduce salt dispersal at least two orders of magnitude to ensure that emission concentrations are well below levels toxic to plant life. Once contaminated, salt-affected soils will require special remedial measures and management practices to restore them to their original productivity.

Potentially, salt would be deposited as dust on the land and would also be transported by runoff to nearby surface waters. Salt concentrations on the order of 8000 parts per million (ppm) are lethal to freshwater fish under conditions of acute exposure (Jones 1964), and the recommended limit for chronic exposure is 80 ppm or 0.01 of the acute toxicity level (NAS 1972). The possibility exists for surface waters, particularly shallow, catch basin-type ponds, to receive amounts of salt sufficient to damage indigenous aquatic plants and animals. Resident species might also be replaced by more salt-tolerant forms.

In addition to effects from dust deposition, localized effects occur from leaching around the surface storage area. Fluctuations in concentrations of soil salinity would depend on precipitation, drainage, seepage, wind and rain erosion rates, and salt concentrations in water and air that come into contact with the soil. Increased salinity around the storage area would decrease or eliminate plant growth, because high salt concentrations in soil reduce the rate at which plants absorb water. This would limit the use of vegetation to increase the aesthetic qualities of the storage area and to control dust.

5.4.4.2 Ecological Effects for a Repository in Granite

A deep geologic radioactive waste repository in granite would be potentially less ecologically damaging than a salt repository and as a consequence would require fewer mitigating measures. During construction, about 8 x 10^7 MT of rock would be mined and 4 x 10^7 MT would require disposal. For convenience of operation the granite would probably be crushed in the mine before being brought to the surface, thereby reducing the airborne dust contamination at the surface.

Possible methods of disposal include removal for use in construction projects (e.g., dams, highways) or surface disposal. Neither of these alternatives pose serious ecological problems. Apart from land use associated with surface storage of the mined material, several hundred tons of airborne particulates may be released yearly. Environmental release of this material to land or surface water could be limited by establishing a vegetation cover for the stored rock, and by proper draining and ponding the surface runoff.

During construction of a granite repository, as with shale and basalt, water may enter either through downward flow from the overlying strata or through upwelling from lower layers. The volume of water entering the repository is generally directly related to repository size and will be greatest during the last stages of construction-operation when the repository is near its maximum size. For granite the estimated inflow of water could be about 1500 m^3/day (400,000 gal/day). Much of this water will be removed as water vapor by the mine ventilation system, although some of the water will probably require collection in sumps in the mine and pumping to the surface. Nonradiological water quality standards will have to be met before this effluent is released to land or surface waters. Disposal of this water will only be necessary until the repository is sealed off. However, the maximum volume of water that would likely need treatment and disposal probably will be less than 760 m^3/day and is not expected to create ecological problems.

5.4.4.3 Ecological Effects for a Repository in Shale

In the case of a deep geological repository in shale about 3.5 x 10^7 MT of rock would be mined and 1.4 x 10^7 MT would require disposal. The mined material would be crushed before it is brought to the surface, a practice that will reduce the release of dust above ground. Several disposal methods may be applicable for mined shale not required for backfilling of the mine. These methods are surface storage, ocean disposal, and placement in abandoned mines. Each of these alternatives has some potential for causing ecological impact. Mine storage may contaminate ground-water supplies that may, in turn, impact ecological systems; some local but poorly defined impacts may result from ocean disposal; and surface storage may remove land from the available natural habitat and be a source of acid runoff.

Shale may contain up to 0.5% iron pyrite, which will produce sulfuric acid when exposed to oxygen and water. Runoff from storage piles, water pumped from the mine, leaching of shale if it were disposed of in abandoned mines, storage, and ocean disposal may provide sources of this acid waste to the environment. The actual quantities and acidity of this waste water have not been defined. Potential ecological impacts will probably be localized and highly site specific. Factors such as the ambient pH of the soil and receiving water, their buffering capacity and the interaction with other physical and chemical parameters will be important in controlling the affects. To afford a moderate level of protection for aquatic life, the pH of freshwater systems should be between pH 6.0 and 9.0, and there should be no change greater than 1.0 units outside the estimated seasonal maximum and minimum (Jones 1964). In marine waters, the addition of foreign material should not reduce the pH below 6.5 or raise it above 8.5, and within the normal range the pH should not vary by more than 0.5 units. Natural plants and animal communities are found on soils ranging from acid bogs to highly alkaline arid environments, and limits of appropriate release would be site specific.

As was the case with the granite repository, shaft and mine liquid effluents are expected to seep into the shale repository during construction. The estimated maximum inflow during the last stages of construction will be about 19,000 m^3/day (5,000,000 gal/day). Most of this water will be collected in sumps, pumped to the surface and treated. One or more holding ponds will be used to retain the water prior to cleanup and release to the environment. Discharge of this volume of water to the environment could require piping or ditching to reduce erosion, and could require sufficient cleanup and neutralization of acid to prevent environmental impact.

5.4.4.4 Ecological Effects for a Repository in Basalt

The expected ecological impacts from the construction and operation of a basalt geologic repository will be small and similar to that of a granite repository. Some impact will occur from noise, dust, and disturbance of surface soil. This will be mainly confined within the 81 ha (200 acre) control zone.

About 9.0×10^7 MT of basalt rock will be mined and 4.4×10^7 MT will require disposal. Suggested disposal methods include surface storage and use in large construction projects (e.g., highways). Several hundred tons of dust will be released per year unless reduced by establishing vegetation on the spoils piles. Erosion through runoff will be controlled by ditching and catch basins. Environmental release of silts from runoff will be small, because the basalt deposits under consideration for a repository are in arid regions. Except for land use considerations, the impacts of the basalt repository will be of little ecological consequence.

5.4.4.5 Ecological Impacts Related to Repositories for Reprocessing Wastes

Ecological effects of repository construction for the reprocessing wastes are expected to be similar to those of spent fuel repositories. Impacts from salt repository construction for these fuel reprocessing wastes are slightly greater than for spent fuel because about 20% more salt is mined. Impacts of granite, shale, and basalt repository construction are less than impacts of spent fuel disposal, because about 32%, 15%, and 34% less materials, respectively, are mined. Again the major ecological impact is from dust depositions that occur from surface handling operations of mined material. Of major concern is the potential for salt depositions at the salt repository fenceline of 11 and 110 g/m^2-yr for the reference and arid environments, respectively.

5.4.5 Nonradiological Accidents

Table 5.4.11 summarizes the number of predicted injuries (temporarily disabling) and fatalities (or permanently disabling injuries) associated with surface facility construction and underground mining operations for the various geologic media for spent fuel and fuel reprocessing waste repositories. These predictions are based on an injury rate of 13.6 temporary disabling injuries per million hours of construction (National Safety Council 1974) for the surface facilities, and an injury rate of 25 temporary disabling injuries per million man-hours for underground mining (other than coal). A fatality rate of 0.17 fatalities (or permanently disabling injuries) per million man-hours of construction (same site) for the surface facilities and 0.53 fatalities per million man-hours for underground mining (other than coal) were used.

Normalizing the construction injuries and fatalities based on standard industrial statistics to a 100,000 MTHM spent fuel repository, the injuries by rock type are about 860,

TABLE 5.4.11. Estimates of Nonradiological Disabling Injuries and Fatalities Associated with Repository Construction Based Upon Current Industrial Statistics for Similar Operations[a]

	Spent Fuel			
	Geologic Media			
	Salt (51,000 MTHM)	Granite (122,000 MTHM)	Shale (64,000 MTHM)	Basalt (122,000 MTHM)
Surface Facility Construction				
• Disabling Injuries	70	70	70	70
• Fatalities	1	1	1	1
Underground Mining Operations				
• Disabling Injuries	370	1400	580	1700
• Fatalities	8	30	12	37
Total				
• Disabling Injuries	440	1500	650	1800
• Fatalities	9	31	13	38
	Fuel Reprocessing Waste			
	Salt (62,000 MTHM)	Granite (69,000 MTHM)	Shale (30,000 MTHM)	Basalt (59,000 MTHM)
Surface Facility Construction				
• Disabling Injuries	84	84	84	84
• Fatalities	1	1	1	1
Underground Mining Operations				
• Disabling Injuries	420	1000	510	1200
• Fatalities	9	21	11	25
Total				
• Disabling Injuries	500	1100	590	1300
• Fatalities	10	22	12	26

(a) Disabling injuries include only temporary disabling injuries; fatalities include permanent disabling injuries.

1200, 1000 and 1500 for salt, granite, shale and basalt, respectively; fatalities amount to about 18, 25, 20, and 31 for salt, granite, shale and basalt, respectively. These losses need to be recognized as perhaps the largest impact associated with the routine management of radioactive wastes, and DOE plans for rigorously enforced safety programs to reduce these potential losses.

5.4.6 Environmental Effects Related to Repository Operation

The operational phase of spent fuel repositories will include the receiving, handling, and placement of spent fuel elements into assigned subterranean storage areas and the subsequent backfilling of these areas when they reach capacity. Similarly, the operational phase of the repositories for reprocessing fuel cycle wastes includes the receiving and handling of wastes, placement of waste canisters and other containers into assigned subterranean storage areas, and the subsequent backfilling of these areas when full.

5.4.6.1 Resource Commitments

Resource commitments for operation of a geologic repository for spent fuel are summarized in Table 5.4.12. Resource commitments for operation of a geologic repository for fuel reprocessing wastes are summarized in Table 5.4.13.

TABLE 5.4.12 Resource Commitments for the Operational Phase of Spent Fuel Geologic Repositories

Materials	Salt (51,000 MTHM)	Granite (122,000 MTHM)	Shale (64,000 MTHM)	Basalt (122,000 MTHM)
PWR canister overpacks, steel, MT	2.5×10^1	5.4×10^1	2.8×10^1	5.4×10^1
BWR canister overpacks, steel, MT	2.8×10^1	6.2×10^1	3.6×10^1	6.2×10^1
PWR retrievability sleeves (5-yr only) steel, MT	8.8×10^3	8.8×10^3	8.8×10^3	8.8×10^3
BWR retrievability sleeves (5-yr only) steel, MT	1.0×10^4	1.4×10^5	1.0×10^4	1.4×10^5
PWR concrete plugs (5-yr only), MT	7.5×10^3	7.5×10^3	7.5×10^3	7.5×10^3
BWR concrete plugs (5-yr only), MT	7.4×10^3	7.4×10^3	7.4×10^3	7.4×10^3
Energy				
Electricity (kWh)	1.5×10^9	3.2×10^9	1.7×10^9	3.2×10^9
Diesel fuel (m^3)	2.1×10^5	3.2×10^5	2.3×10^5	3.2×10^5
Coal (MT)	1.2×10^6	1.8×10^6	1.3×10^6	1.8×10^6
Manpower (man-years)	1.1×10^4	2.0×10^4	1.3×10^4	1.9×10^4

5.4.6.2 Nonradiological Effluents

The major nonradiological effluent from facility operation would be fugitive dust emissions from surface handling of mined materials, as was discussed under construction impacts (Section 5.4.4). Other nonradiological pollutants released to the biosphere during the repository's operational life are given in Tables 5.4.14 and 5.4.15 for the various geologic media. These pollutants include combustion products from burning diesel fuel (URS 1977) during underground mining operations and from surface burning of coal (OWI 1978).

The estimated releases of pollutants from a geologic repository as given in Table 5.4.14 would not, in any case, result in Federal Air Quality Standards being exceeded at the repository boundary. For example, the maximum concentration of particulates at the repository boundary (1.6 km from point of release, where the $\overline{X}/Q'$ is 1×10^{-6} sec/m^3) was estimated to be 0.8 $\mu g/m^3$ compared to the standard of 75 $\mu g/m^3$.

Heat released from buried nuclear waste will increase the temperature of the geologic formation in which it is buried and may alter the physical and chemical properties of the

TABLE 5.4.13. Resource Commitments for the Operational Phase of Fuel Reprocessing Waste Geologic Repositories

Materials	Salt (62,000 MTHM)	Granite (69,000 MTHM)	Shale (30,000 MTHM)	Basalt (56,000 MTHM)
HLW canister overpacks, MT steel[a,b]	6.4	8.2	4.8	9.0
RH-TRU canister overpacks, MT steel	1.5×10^1	1.6×10^1	1.0×10^1	1.4×10^1
RH-TRU drum packs, MT steel	5.3×10^4	5.8×10^4	3.0×10^4	4.9×10^4
HLW retrievability sleeves, MT steel[b,c]	7.3×10^2	9.6×10^2	1.3×10^3	1.3×10^3
RH-TRU retrievability sleeves, MT steel[c]	2.9×10^4	1.9×10^5	2.9×10^4	1.6×10^5
HLW concrete plug,[c] MT	8.0×10^2	1.0×10^3	1.4×10^3	1.4×10^3
RH-TRU concrete plug, MT concrete[c]	7.2×10^4	7.2×10^4	7.2×10^4	7.2×10^4
Energy				
Electricity, kWh	2.1×10^9	2.6×10^9	1.4×10^9	2.3×10^9
Coal, MT	1.4×10^6	1.4×10^6	9.4×10^5	1.3×10^6
Diesel fuel, m^3	2.5×10^5	2.6×10^5	1.7×10^5	2.3×10^5
Steam, MT	1.5×10^7	1.6×10^7	1.0×10^7	1.4×10^7
Manpower, man-yr	1.9×10^4	2.4×10^4	1.3×10^4	2.1×10^4

(a) Overpack requirements are based on 0.1% of canisters received leaking or damaged.
(b) HLW canister and sleeve diameters change with time as necessary to maintain canister heat output within limits.
(c) Sleeves and plugs needed for first five years only.

TABLE 5.4.14. Total Quantities of Effluents Released to the Atmosphere During Operation of a Geologic Repository for Spent Fuel

Effluent	Geologic Medium: Salt	Granite	Shale	Basalt
Particulates, MT	430	670	480	670
SO_x, MT	9,700	15,000	11,000	15,000
CO, MT	2,400	3,700	2,700	3,700
Hydrocarbons, MT	870	1,400	980	1,400
NO_x, MT	15,000	24,000	17,000	24,000
Heat, MJ	3.9×10^8	9.3×10^8	4.9×10^8	9.3×10^8

formation. The heat will eventually be transferred to the atmosphere and, if the temperatures and temperature gradients have not exceeded values that would cause damage to the formation or adversely affect the containment integrity or the environment, the formation will return essentially to its initial state. The maximum surface temperature increase in any case is not expected to exceed about 0.5°C. This aspect is discussed more fully in Section 5.5 and in DOE/ET-0029.

TABLE 5.4.15 Total Quantities of Effluents Released to the Atmosphere During Operation of Geologic Repository for Reprocessing Wastes

Effluent	Geologic Medium			
	Salt	Granite	Shale	Basalt
Particulates, MT	510	540	350	480
SO_x, MT	12,000	12,000	7,800	11,000
CO, MT	2,900	3,000	2,000	2,700
Hydrocarbons, MT	1,000	1,100	710	980
NO_x, MT	17,000	19,000	12,000	17,000
Heat, MJ	7.6×10^8	8.3×10^8	4.3×10^8	7.0×10^8

5.4.6.3 Radiological Releases

Routine radiological releases from geologic repositories during normal operation will consist principally of radon emanating from exposed rock faces and radon's decay products. These releases will also occur from backfilling operations but are negligible compared to radon releases during repository construction. Occasionally, external contamination may occur on canisters as a result of some minor accident. The population dose from decontamination activities would be much less than that from operation at a spent fuel packaging and storing facility, for which the 70-yr whole-body population dose was determined to be about 1 man-rem (DOE/ET-0029).

Doses to the work force during repository operation will include contributions from receiving, handling, and placement of waste canisters into subterranean storage areas. Doses estimated to result from operations, based on expected time of operation and permissible exposure limits, are presented below for disposal of wastes for the various geologic media:

Geologic Media	70-Year Whole-Body Dose (man-rem)	
	Spent Fuel Repository	Reprocessing Waste Repository
Salt	4.3×10^3	1.4×10^5
Granite	1.1×10^4	1.6×10^5
Shale	5.6×10^3	8.0×10^4
Basalt	1.1×10^4	1.3×10^5

Radiation-related health effects using the conversion factor of 100 to 800 health effects per million man-rem (Appendix E) suggests a range of zero to 130 health effects among a workforce of about 8000. The doses tabulated suggest individual worker doses of about 1 rem per year over a 15-year repository loading period.

5.4.6.4 Ecological Impacts

The major ecological impact of repository operation would be from the handling of mined materials at the surface during repository mining and backfilling. Impacts would be caused by the airborne transfer of mined particulates to the environment near the site. These impacts would be greatest for the repository in salt. Mitigating procedures may be necessary to control this potential threat to the environment. Impacts of fugitive dust were discussed in Section 5.4.4.

5.4.6.5 Socioeconomic Impacts

Socioeconomic impacts associated with the construction and operation of repositories are dependent largely on the number of persons who move into the locality in which the facility will be located. Because of this, the size of the local project-generated population influx was forecasted, and estimates of their needs for locally provided social services were determined. Specific economic and fiscal impacts attributable to the development of the repository cannot be treated here because they are too site dependent

Socioeconomic impacts also depend on site characteristics (see DOE/ET-0029, Appendix C) and the assumptions used for forecasting. Site characteristics that are especially important in influencing the size of the impacts include the availability of a skilled local labor force, secondary employment, proximity to a metropolitan area, and demographic diversity (population size, degree of urbanization, etc.) of counties in the commuting region. An additional factor in the generation of impacts is the time pattern of project-associated population change. For example, a large labor force buildup followed closely by rapidly declining project employment demand could cause serious economic and social disruptions near the site and elsewhere within the commuting region.

Impacts are estimated for three reference sites, identified as Southeast, Midwest, and Southwest (see Appendix G). These areas were chosen because they differ substantially in demographic characteristics, thus providing a reasonable range of socioeconomic impacts.

The socioeconomic model employed in this analysis first forecasts a regional population in 5-yr intervals in the absence of any project activities. This population forecast serves both as a comparative baseline and as a source for a portion of the postulated future project employment. The model takes into account both primary (project related) and secondary employment effects (such as additional retail store clerks) and incorporates as separate components spouses of members of the labor force and other dependents. Projected residences of regional migrants associated with the project are distributed to counties throughout the commuting region. The model accounts for separation and retirement from project employment and replacement by new labor force members. It also accounts for the tendency of workers and their dependents to leave the region upon job separation.

In the following analysis, impacts are presented in terms of an expected level of impact. Maximum levels of impact were also calculated and appear in DOE/ET-0029. The expected impact condition is based on the most likely value of model assumptions, whereas the maximum impact condition places an extreme but credible value on the model assumption.

Table 5.4.16 presents the manpower requirements for construction and operation of a single waste repository involving spent fuel or reprocessing of wastes.

Table 5.4.17 presents estimates of the cumulative project-related in-migrants for the three reference repository sites in salt. Similar estimates were made for granite, shale,

TABLE 5.4.16 Estimated Manpower Requirements for Construction and Operation of a Single Waste Repository, by Disposal Average Annual Employment (3-yr. peak)

	Spent Fuel Repository		Reprocessing Waste Repository	
Medium	Construction	Operation	Construction	Operation
Salt	1700	870	2000	1300
Granite	4200	1100	3000	1300
Shale	2200	880	2100	1200
Basalt	5000	1100	3800	1500

TABLE 5.4.17. Forecasts of Expected Population Influx for a Geologic Repository in Salt (51,000 MTHM Waste Capacity): Number of Persons and Percent of Base Population[(a)]

	Site	1980	1985	2000	2005
Spent Fuel Repository	Southeast	330 (1.9%)	540 (3.0%)	660 (3.3%)	700 (3.4%)
	Midwest	130 (0.2%)	570 (0.8%)	710 (0.9%)	740 (0.9%)
	Southwest	5,200 (10.8%)	4,200 (8.5%)	5,000 (9.2%)	5,100 (9.1%)
Reprocessing Waste Repository	Southeast	410 (2.3%)	760 (4.1%)	930 (4.6%)	980 (4.7%)
	Midwest	200 (0.4%)	860 (1.3%)	1,100 (1.3%)	1,100 (1.3%)
	Southwest	6,200 (12.4%)	5,700 (11.3%)	6,800 (12.1%)	6,900 (12.0%)

(a) The dates shown are for one possible scenario and do not attempt to reflect actual schedules. The effects of population influx are expected to be substantially the same regardless of actual startup date.

and basalt and are presented in DOE/ET-0029. The forecasted values include primary and secondary workers and associated household dependents, all of whom are in-migrants. Some of the persons who separate from the facility will stay in the site county and some will leave. Those who will stay are included in the forecasted values. Thus, not all forecasted populations are actually working on or directly associated with the project at each time period. Nevertheless, the presence of each of these persons would be caused by the existence of the project; they would probably not be present if the project did not occur. The percentages associated with each population in these tables reflect the size of the in-migrant group relative to the baseline population in the respective sites. Since these baseline populations vary by site, the relative impact of a similar in-migrant group can vary greatly.

Manpower requirements for construction of disposal facilities are lowest for a repository in salt and highest for a repository in basalt. For a spent fuel repository in salt, the total numbers of forecasted new in-migrants in the Southeast and Midwest sites under expected impact conditions are under 3% of the site county populations in the construction (1980-1984) and operation (1985-2005) phases. In-migration at this level is not likely to produce significant impacts. The effect of a repository in salt at the Southwest site is substantially different. The number of in-migrants during construction is over three times the level of primary employment demand (4200 versus 1700). Project related in-migration that exceeds 10% of the corresponding baseline population is considered to produce significant impacts. In-migration to the Southwest site exceeds this level in most cases. For a repository in granite, expected impacts at the Southeast and Midwest sites are judged to be non-significant. Again, the Southwest site is subjected to relatively large impacts, primarily because there is a scarcity of skilled available local labor.

The translation of forecasted project-related in-migration into socioeconomic impacts is complex and imprecise. Estimates of the level of demand that will be placed on the community to provide social services to the new workers and their families were made by applying a set of factors (see DOE/ET-0029, Appendix C) to the project in-migration values. The product indicates how many units of each social service would be "expected" by the in-migrants. The severity of these impacts is primarily related to the capacity of the site county to adsorb these expected values. To contain all of the spent fuel in a 10,000 GWe-yr scenario, eight reference repositories in salt, three in granite or basalt, or six in shale were estimated to be required; thus, the impacts described would occur 8, 3, or 6 times (but in different places) depending on the medium chosen for disposal. In a similar way the impacts for construction of fuel reprocessing waste repositories would occur 6, 7 or 10 times depending on media chosen for disposal. (See Chapter 7 for numbers of repositories required in different power growth scenarios.)

The calculated level of the expected need for additional social services at the three reference sites is given for the year 2000 for spent fuel and fuel reprocessing repositories in Tables 5.4.18 through 5.4.21. Identification of social services that would likely be required indicates the potential extent of socioeconomic impacts. The ability of communities to provide services identified here, with or without financial assistance, is highly site-specific and is beyond the scope of this document. Some of the social services listed can be described as operational, such as physicians and teachers. These needs are more easily met on a temporary, less-costly basis than are those services that require major capital investment. The latter include hospital beds to the extent that hospital space is also needed, classroom space, and additional sanitary waste treatment capacity. Capital investment needs are forecast to be large, especially in the Southwest site, and to the extent that they persist over time, they will represent a serious challenge to community planners and local government. The increase in the local crime rate is only one indicator of the social disruption and a sense of a decline in social well-being experienced by community residents faced with large-scale development. This analysis does not address one site-specific but very important impact of any major construction activity; that is the impact of increased property values, increased taxes and increased commodity prices on fixed-income families.

In general, the reference Southwest site is more likely to sustain significant socioeconomic impacts compared with the other two sites, because it has a smaller available unemployed construction labor force, lacks a nearby metropolitan center, and is subject to the generation of greater secondary employment growth compared with the other sites. If a repository were to be built in an area where demographic conditions approximated that of the Southwest site, a detailed analysis of site-specific socioeconomic impacts would be needed to help prevent serious disruptions in provision of necessary social services.

TABLE 5.4.18. Selected Expected Social Service Demands Associated with Migration into the Site County Resulting from the Construction and Operation of a Geologic Repository in Salt

Selected Social Services	Year 2000					
	Spent Fuel Repository			Reprocessing Waste Repository		
	Southeast Site	Midwest Site	Southwest Site	Southeast Site	Midwest Site	Southwest Site
Health						
Physicians and dentists	1	1	5	1	2	7
Hospital and nursing care beds	3	5	20	4	8	28
Education						
Teachers	8	8	66	11	13	90
Classroom space, m^2	760	790	7,300	1,100	1,200	9,900
Sanitation						
Water treatment, m^3/d	300	330	2,400	430	490	3,200
Liquid waste, m^3/d	200	220	1,600	290	320	2,200
Fire and police, personnel	2	2	11	2	2	15
Recreation areas, ha	1	1	5	1	1	7
Government						
Administrative staff	1	1	4	1	1	5
Other social impacts						
Crimes (7 crime index)	25	25	240	35	40	330

TABLE 5.4.19. Selected Expected Social Service Demands Associated with Migration into the Site County Resulting from the Construction and Operation of a Geologic Repository in Granite

Selected Social Services	Year 2000					
	Spent Fuel Repository			Reprocessing Waste Repository		
	Southeast Site	Midwest Site	Southwest Site	Southeast Site	Midwest Site	Southwest Site
Health						
Physicians and dentists	1	3	9	1	2	9
Hospital and nursing care beds	4	11	35	5	11	35
Education						
Teachers	13	18	117	14	17	111
Classroom space, m^2	1,200	1,500	13,400	1,300	1,500	13,000
Sanitation						
Water treatment, m^3/d	510	800	4,200	530	730	4,200
Liquid waste, m^3/d	340	530	2,800	350	490	2,800
Fire and police, personnel	2	3	20	3	4	20
Recreation areas, ha	1	2	9	1	2	9
Government						
Administrative staff	1	1	7	1	1	7
Other social impacts						
Crimes (7 crime index)	40	60	430	40	60	430

TABLE 5.4.20. Selected Expected Social Service Demands Associated with Migration into the Site County Resulting from the Construction and Operation of a Geologic Repository in Shale

	Year 2000					
	Spent Fuel Repository			Reprocessing Waste Repository		
Selected Social Services	Southeast Site	Midwest Site	Southwest Site	Southeast Site	Midwest Site	Southwest Site
Health						
Physicians and dentists	1	1	6	1	2	7
Hospital and nursing care beds	3	6	22	4	8	27
Education						
Teachers	9	10	74	11	12	89
Classroom space, m^2	820	910	8,300	1,000	1,100	9,800
Sanitation						
Water treatment, m^3/d	330	400	2,700	410	490	3,200
Liquid waste, m^3/d	220	270	1,800	280	320	2,100
Fire and police, personnel	2	2	13	2	2	15
Recreation areas, ha	1	1	6	1	1	7
Government						
Administrative staff	1	1	4	1	1	5
Other social impacts						
Crimes (7 crime index)	30	30	280	30	40	330

TABLE 5.4.21. Selected Expected Social Service Demands Associated with Migration into the Site County Resulting from the Construction and Operation of a Geologic Repository in Basalt

	Year 2000					
	Spent Fuel Repository			Reprocessing Waste Repository		
Selected Social Services	Southeast Site	Midwest Site	Southwest Site	Southeast Site	Midwest Site	Southwest Site
Health						
Physicians and dentists	1	3	11	1	3	11
Hospital and nursing care beds	5	13	39	5	13	50
Education						
Teachers	14	21	130	15	20	132
Classroom space, m^2	1,300	1,700	15,000	1,400	1,800	14,800
Sanitation						
Water treatment, m^3/d	550	930	4,700	600	860	4,700
Liquid waste, m^3/d	370	620	3,100	400	570	3,200
Fire and police, personnel	3	4	22	3	4	22
Recreation areas, ha	1	2	10	1	2	10
Government						
Administrative staff	1	2	7	1	1	8
Other social impacts						
Crimes (7 crime index)	45	70	480	50	65	490

5.4.6.6 Environmental Effects Related to Postulated Radiological Accidents

Several accidents that could result in the release of radionuclides were analyzed for the spent fuel repositories. The accidents were chosen on the basis of their probability of occurence and radiological consequences. Of accidents which might occur during the operation phase, the drop of a spent fuel canister down the repository mine shaft was most serious and its effects are presented here. Severe accidents after repository closure are treated in Section 5.5. Scenarios are provided in DOE/ET-0028.

For the accident involving a canister dropped down a repository mine shaft, radionuclides are assumed to be released to the mine atmosphere from the failed canister over a period of 1 hr. An elevator load is assumed to include four spent fuel assemblies containing 2 MTHM of spent fuel that are assumed to be ten years out of the reactor. The radioactive materials that would be released to the environment from such an accident are presented in Table 5.4.22. The releases were determined using the assumption that material released in the mine shaft passes through a roughing filter and two HEPA filters (total Decontamination Factor (DF) for particulates of 10^7) prior to release to the environment through a 110-m stack. Frequency of occurrence of the accident is postulated to be 1×10^{-5} per year.

Based on these releases, the 70-yr whole-body dose commitment to the maximum individual[a] was calculated to be 3.5×10^{-5} rem. The 70-yr whole-body doses to the world-wide population would be 8.7 man-rem, compared with 4.5×10^{10} man-rem from naturally occurring sources.

Accidents were also postulated for the geologic repository for reprocessing wastes that might lead to release of radionuclides to the environs and are listed in Table 5.4.23. Scenarios are provided in Section 7.3.1.9 of DOE/ET-0028 and analyses of the accidents are presented in DOE/ET-0029. Non-design-basis accidents are discussed in Section 5.5.

Of the minor accidents, the contact-handled transuranic (CH-TRU) waste drum rupture accident (handling error) was considered most representative of the minor accidents. In this minor accident, a forklift operator error is assumed to result in the breach of one drum of CH-TRU waste. The accident can occur in the surface facility or in the CH-TRU waste mine shaft and has an estimated frequency of 0.15/yr. For the 0.63 MTHM equivalent contained in a single drum, a release fraction of 2.5×10^{-5} over a release time of 30 minutes was used.

Radioactive materials that would be released to the outside environment from this accident are presented in Table 5.4.24. The releases are assumed to be the same whether the accident occurs in the surface facility or the CH-TRU waste mine, since all releases would be released from a mine exhaust stack approximately 100 m high.

(a) The maximum individual is defined as a permanent resident at a location 1600 m southeast of the stack with the time-integrated atmospheric dispersion factor (E/Q) of 1.3×10^{-5} sec/m^3.

TABLE 5.4.22. Radioactive Material Released to the Atmosphere from a Spent Fuel Canister Drop-Down-Mine-Shaft Accident at a Geologic Repository

Radionuclide	Ci	Radionuclide	Ci
^{3}H	6	^{238}Pu	4.0×10^{-6}
^{14}C	4×10^{-2}	^{239}Pu	5.8×10^{-7}
^{85}Kr	4×10^{3}	^{240}Pu	9.0×10^{-7}
^{90}Sr	1.0×10^{-4}	^{241}Pu	1.4×10^{-4}
^{90}Y	1.0×10^{-4}	^{241}Am	3.2×10^{-6}
^{129}I	6×10^{-3}	^{244}Cm	1.8×10^{-6}
^{137}Cs	1.5×10^{-4}		

TABLE 5.4.23. Postulated Accidents for the Geologic Repository for Reprocessing Wastes

	Accident Number	Accident
Minor	7.1	CH-TRU transuranic waste drum rupture caused by a handling error
	7.2	Minor canister failure due to rough handling
	7.3	Externally contaminated canister
	7.4	Receipt of dropped shipping cask
Moderate	7.5	Canister drop in surface facility
	7.6	Canister drop down mine shaft
	7.7	Tornado strikes salt storage piles
	7.8	CH-TRU waste drum rupture caused by mechanical damage and fire
	7.9	CH-TRU waste drum rupture caused by internal explosion

TABLE 5.4.24. Radioactive Material Released to the Atmosphere from a CH-TRU Waste Accident at the Geologic Repository for Reprocessing Wastes, Ci

Nuclide	U and Pu Recycle	Nuclide	U and Pu Recycle
^{3}H	6.3×10^{-6}	^{129}I	1.6×10^{-9}
^{14}C	1.6×10^{-10}	^{134}Cs	1.8×10^{-12}
^{60}Co	6.2×10^{-13}	^{137}Cs	1.4×10^{-12}
^{90}Sr	9.2×10^{-13}	^{238}Pu	8.2×10^{-12}
^{95}Nb	1.1×10^{-11}	^{239}Pu	5.4×10^{-13}
^{106}Ru	2.8×10^{-10}	^{240}Pu	1.1×10^{-12}
		^{241}Pu	2.7×10^{-10}

Based on the CH-TRU releases listed in Table 5.4.24, the 70-yr dose commitment to the maximum individual was calculated to be 1.0×10^{-12} rem, which is a number so small as to be effectively zero. For the same period, the maximum individual would receive about 7.0 rem from naturally occurring sources.

The 70-yr worldwide population dose from ^{3}H and ^{14}C calculated for this case is approximately 3.9×10^{-18} man-rem, which is effectively zero when compared with 4.5×10^{10} man-rem received from naturally occurring sources.

Calculations of the effect of a drop of a fuel reprocessing waste canister down the mine shaft indicated that this would be categorized as a moderate accident in terms of release outside the repository. Some of the canistered waste is assumed to be released to the mine atmosphere from four failed canisters in a time period of 1 hour. Canistered waste will be one of three forms:

- Solidified High-Level Wastes:
  - Glass (175 kg/MTHM)--13 kg of particles less than 10 m in diameter will be released to mine filters. Postulated frequency of occurrence is 7×10^{-7}/yr.
  - Calcine (52.5 kg/MTHM)--31 kg of particles less than 10 m will be released to mine filters. Frequency of occurrence is 7×10^{-7}/yr.
- RH-TRU Wastes--1.3 kg of Zircaloy fines less than 10 m in diameter will reach the mine filters. The postulated frequency of occurrence is 2×10^{-6}/yr.

The radioactive materials that would be released to the outside environment for the various waste forms are presented in Tables 5.4.25. These releases were calculated assuming that material released in the mine shaft passes through a roughing filter and two HEPA filters (DF of 10^{7}) prior to escaping to the environment through a 110-m stack.

Doses to the maximum individual from these accidents are given in Table 5.4.26. The doses in Table 5.4.26 are insignificant in terms of the radiation dose of 7 rem the individual would have received from naturally occuring sources over the same time period.

TABLE 5.4.25. Radionuclide Releases for a Waste Canister Dropped Down a Mine Shaft at a Repository for Reprocessing Wastes, Ci

Nuclide	Glass	Calcine	Nuclide	RH-TRU
^{90}Y	3.9×10^{-4}	3.2×10^{-3}	^{3}H	2.5×10^{-1}
^{90}Sr	3.9×10^{-4}	3.2×10^{-3}	^{14}C	4.4×10^{-4}
^{106}Ru	4.4×10^{-5}	3.4×10^{-4}	^{60}Co	1.6×10^{-6}
^{125m}Te	4.8×10^{-6}	4.0×10^{-5}	^{63}Ni	1.6×10^{-7}
^{134}Cs	8.0×10^{-5}	1.3×10^{-3}	^{90}Sr	1.2×10^{-8}
^{137}Cs	6.0×10^{-4}	4.8×10^{-3}	^{54}Mn	8.1×10^{-8}
^{144}Ce	2.0×10^{-5}	1.6×10^{-4}	^{95}Nb	8.2×10^{-8}
^{154}Eu	3.6×10^{-5}	2.8×10^{-4}	^{137}Cs	1.9×10^{-8}
^{238}Pu	5.6×10^{-7}	4.4×10^{-6}	^{144}Ce	4.8×10^{-8}
^{239}Pu	1.3×10^{-8}	1.0×10^{-7}	^{238}Pu	1.1×10^{-9}
^{240}Pu	5.2×10^{-8}	4.0×10^{-7}	^{239}Pu	7.2×10^{-11}
^{241}Pu	6.4×10^{-6}	4.0×10^{-5}	^{240}Pu	1.5×10^{-10}
^{241}Am	5.2×10^{-6}	4.0×10^{-5}	^{241}Pu	3.6×10^{-8}
^{244}Cm	4.4×10^{-5}	3.5×10^{-4}	^{241}Am	1.4×10^{-10}
			^{242}Cm	2.0×10^{-9}
			^{244}Cm	1.4×10^{-9}

TABLE 5.4.26. 70-Yr Whole-Body Dose Commitments to Maximum Individual from Drop of Waste Canisters into a Geologic Repository

Waste	70-Yr Dose Commitment, rem
High-Level	
Calcine	1.2×10^{-4}
Glass	1.4×10^{-5}
RH-TRU	1.7×10^{-7}

In summary, radiological aspects of repository construction and routine operation including reasonably forseeable accidents while filling and decommisioning the repository do not constitute a significant impact on public health and safety.

5.4.6.7 Radiological Impacts of Operating Accidents on the Work Force

In the case of reprocessing waste, the calculated first-year total-body dose to a member of the repository work force near the point of impact of four canisters of high-level waste dropped down a mine shaft would be 26,000 rem for waste in glass, about 210,000 rem for the waste in calcine form, and about 7,600 rem for the spent fuel case; all fatal doses.[a] The exposure rate in the corridor due to contamination of surfaces would be approximately 20 R per hour from the waste (about 5 R per hour in the case of spent fuel). Such exposure rates would make decontaminating the corridor impossible by ordinary means; some sort of remote operation similar to that of dismantling a reactor core would be needed. However, design changes to the transfer stations in the repository and the use of two stages of HEPA filtration between the shaft and other portions of the mined repository would probably lower the occupational doses to repository workers to within acceptable ranges. These changes would limit the area contaminated to the transfer station and possibly the canistered waste (CW) shaft; although air flow should preclude significant contamination in the CW shaft. Limiting the contaminated area should also decrease the time required for decontamination and resumption of repository loading.

5.4.6.8 Other Environmental Impacts

An artist's rendering, based on engineering data, of the above-ground facilities associated with a geologic repository was shown in Figure 5.3.1. With the exception of the mine spoils piles, these facilities would not be expected to be any more of a detraction than any other mining or industrial facility of comparable size. Although the exclusion boundary could be viewed as a detraction in itself, the exclusion area will likely limit the visual impacts of the above-ground repository facilities.

(a) The source terms used in these calculations are believed to be unrealistically pessimistic but additional engineering analysis is necessary before the source terms can be reduced with confidence.

The spoils piles could have an adverse visual impact. If left onsite, these spoils, if piled 3 meters high (about 10 feet), would occupy about 2 to 5 km^2 ($\sim$1 to 2 square miles). This amount of material is equivalent to 13 to 44 million tons of rock, depending on repository host rock, and might be used in the construction of markers for the repository.

In the case of repositories in salt, little noise other than that from traffic would be expected in conjunction with repository construction. In the case of shale repositories construction would probably be performed with occasional blasting when encountering tightly bound hard portions of the rock; otherwise, as in the case of salt, little noise would be discernible at the surface. In the case of basalt and granite, almost all rock removal will require blasting and, as a consequence, considerable blast noise or, more likely, ground rumble would result. The degree of annoyance produced would depend in large part on the proximity of populated areas to the repository.

There were no identifiable sources of odor unique to the construction and operation of a geologic repository for radioactive waste. Increased air pollution from construction and commuter vehicles is expected; however, this is not expected to be experienced as odor. Stacks at the coal-fired support plants will be designed to mitigate noxious odors and ash from coal burning.

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5.5 LONG-TERM ENVIRONMENTAL CONSIDERATIONS OF GEOLOGIC DISPOSAL OF RADIOACTIVE WASTES[(a)]

The objective of disposal of radioactive wastes in deep geologic repositories is to provide reasonable assurance[(b)] that the radionuclides contained in these wastes in biologically significant concentrations will be permanently isolated from the human environment. The following presentation examines the likelihood and consequences of events that could compromise this objective over the millenia following repository closure.

No significant long-term physical impacts are expected to result from having placed the heat-emitting radioactive wastes in geologic repositories as described previously in this Statement whether located in salt, granite, shale or basalt formations. Although heat from decaying radionuclides will ultimately reach the surface of the earth via conduction through overlying rock, temperature rises at the surface were estimated to be less than 0.5°C in all cases. Such a temperature rise is insignificant. Heat flowing into and through the rock surrounding repositories will cause expansion of the rock and would result in some uplift at the surface. The largest uplifts (over several centuries) are expected to be on the order of 0.3 to 0.6 m (1 to 2 ft) in shale, and 1.2 to 1.5 m (4 to 5 ft) in salt at the center of the 800 ha (2000 acre) repository area.

Subsidence of the formation containing a waste repository following closure or collapse of the void spaces that remained after the mine has been backfilled (backfilled to 60% of volume) might occur at repositories in salt and shale. Uplift and subsidence are expected to occur over very long time periods, and as a consequence no impacts associated with earth movement are expected to result. For repositories located in granite and basalt, subsidence or uplift is believed unlikely.

Nuclear waste repositories will be sited, loaded, and sealed with every expectation that long-term radiological impacts will be nonexistent. There are, however, a few highly improbable events that can be postulated to take place singularly (or in combination with smaller probability events) that might result in radioactive wastes reaching the biosphere. Three kinds of events leading to release of some of the repository contents were postulated:

- direct release of contents to the atmosphere: Such release could follow volcanic activity, impact of a large meteorite or large nuclear weapon, or, on a much longer time scale, denuding of the earth to the depth of the repository by erosion or glaciation. Releases and consequences of these events are believed to be adequately represented by those of a meteorite strike; however, the probability of occurrence could be substantially different.

(a) "Long-term" as used here means hundreds to tens-of-thousands of years after the repository has been closed.
(b) "Reasonable assurance" is admittedly a subjective expression. While DOE believes that shallow land burial for spent fuel, HLW, remotely handled TRU or fuel reprocessing wastes would not give such reasonable assurance, DOE believes that at some depth isolation is reasonably assured. Depths on the order of hundreds of meters are believed to meet this requirement.

- release via water: Water might enter a repository as a result of flooding or seepage following the breach of overlying rock by such mechanisms as fracturing by faulting, nearby impact of meteorite or nuclear weapon, thermal stresses caused by decay heat from the radioactive waste, mechanical stresses resulting from adjustment of repository rock following excavations, or failure of shaft and/or bore hole seals. Plausible events can be postulated whereby water enters even a well-sited repository; far less plausible are events that would bring the potentially contaminated water back to the surface or to aquifers reasonably penetrable by wells.
- release via man-made intrusions: These might include exploratory drilling, solution mining of salt or phosphates; or cavern construction for storage of oil, industrial wastes, compressed air, etc.

Several of these events were chosen to provide a basis for estimating the risk of waste disposal to society. Events representative of the above categories are:

- meteorite impact penetrating to the waste bearing stratum[a]
- fracturing through rock overlying the repository by faulting followed by stream flooding or slow groundwater infusion
- exploratory drilling through a waste canister
- solution mining for salt content, in the case of a repository in salt.

The event analyses that follow are based on the concept of "what if" they occur. In cases where probabilities could be assigned, they were used to provide an estimate of societal risk from the disposal of radioactive waste in deep geologic repositories. Following each accident discussion, a description of any action that may be taken to mitigate the consequences of the accident is presented.

Modeling methods used to estimate the consequences of the accidents are described in the appendices: Appendix D, Models Used in the Dose Calculations; and Appendix E, Radiologically Related Health Effects. Methods not described in the appendices are referenced in the text.

The radiological release consequences of the meteorite and faulting and flooding (ground-water transport) accidents are based on the assumption that breaches occur in the repository host rock itself and consequently, differing properties of the different host rocks do not enter into the calculation of the consequences. Therefore, the differences in consequences in terms of repository media are inventory related; results differ only because of the different amounts of waste disposed of in each repository. The amounts in the repositories were developed on the basis of waste emplacement in 800 ha per repository and are as follows:

(a) Representative in the sense of release and consequence but not necessarily in the sense of probability of occurrence.

	Spent Fuel	Fuel Reprocessing Waste HLW	Fuel Reprocessing Waste RH & CH-TRU
Salt	51,000 MTHM[(a)]	62,000 MTHM	100,000 MTHM
Granite	122,000	69,000	108,000
Shale	64,000	30,000	56,000
Basalt	122,000	56,000	92,000

If the amount of disposed waste, rather than the size of the repository, were held constant, the radiological consequences would be the same for each geologic medium. In other words, once the radionuclides are outside the repository proper, their movement away from the repository is governed by the same set of assumptions regardless of repository media. (This limitation of the analysis would be improved upon in site-specific analyses when site specific data or sorptive properties of adjacent rock become available.)

In the case of faulting and flooding with stream transport the assumption was made that the same amount of waste was removed by water regardless of repository medium. Repository medium affected consequences only in salt repositories; the presence of salt along with the wastes would likely preclude use of the emergent stream as a source of drinking water or food. Thus, except for the case of salt entering the biosphere with the waste radionuclides, no analysis was made of the waste repository medium's influence in the consequences of the postulated long-term events.

In the case of human intrusion by drilling, the same amount of waste was assumed to be brought to the surface regardless of repository media.

5.5.1 Repository Breach by Meteorite

Breach of a repository would be possible by a meteorite estimated to be about 25 m in diameter striking a point on the surface above the center of the repository at a speed of about 20 km/sec on impact. If the meteorite's density is 8 g/cm^3 (which is representative of iron or nickel-iron meteorites), the mass of the meteorite at contact would be about 6.5 x 10^4 MT and would have an energy equivalent to about 3 megatons of TNT. This meteorite would produce a crater roughly 2 km in diameter at the surface and 600 m deep at its deepest point. No clear evidence is available to suggest that meteorites of this size have created craters this deep over the age of the earth. On the other hand, the presence of astroblemes suggests that the earth has been hit by very large extraterrestrial bodies (Claiborne and Gera, 1978).

Temperatures at the impact point of the meteorite strike would reach millions of degrees, and most of the meteorite plus some of the surrounding rock would be vaporized. Some of the rock material would be pulverized and ejected into the air as the crater formed. Most of the ejected material would fall back into the crater and its immediate vicinity.

(a) Metric tons of heavy metal in the case of spent fuel or spent fuel equivalent in the case of reprocessing wastes.

If the meteorite had an energy equivalent of about 3 megatons of TNT, the overall effects would be somewhat like those from a nuclear weapon but without the prompt radiation effect.[a] Thus, a shock wave as well as thermal effects could be expected. If a 3-megaton nuclear weapon were detonated, any individual residing within 4 km from the point of impact would be killed or would suffer at least second-degree burns and other injuries from the blast, falling buildings, and flying debris, etc.

Radioactive material suspended by a meteorite impact would be dispersed by two modes, developed on the basis of nuclear cratering test results. A typical cloud formation consists of a central column rising about a doughnut-shaped base surge, which rolls outward from the crater. One-half of the suspended material is dispersed in the central column and one-half is dispersed in the base cloud. For the reference midwest site, the material in the central cloud is also dispersed evenly across the eastern half of the United States and then moved around the world at high altitude. Compared to the base cloud, it does not contribute significantly to local (radius of 80 km) fallout. Because of large overpressures in air produced on impact of the meteorite, local low-altitude winds are assumed to have no affect on dispersion of material.

If the meteorite impact penetrated to a depth of 600 m, the impact is arbitrarily assumed to result in dispersion of about 1% of the repository inventory. The amounts of various radionuclides ejected depend on the length of time between repository closure and meteorite impact. This event was examined for a meteorite strike at the assumed time of repository closure (therefore maximum waste disposal inventory) and for 1000, 100,000 or 1,000,000 years thereafter. Assumptions about dispersion of radioactive material after meteorite impact are summarized below.

Ten percent of the particulate radioactive material dispersed is assumed to be of respirable size (^{3}H, ^{14}C, ^{85}Kr, and ^{129}I are assumed to be released as gases and all other radionuclides are assumed to be in particulate form). The remaining 90% of the particulate material falls back immediately into or near the crater and does not contribute to the regional population dose. For calculation of the dose to the regional population, the amount dispersed is also reduced by an additional one-half to account for the distribution of material between central and base clouds.

First-year and 70-year cumulative doses to the whole-body for various times of repository breach and for repositories in various media are presented in Tables 5.5.1 and 5.5.2. Doses to individual organs, a breakdown of dose by pathway, and tabulations of the radionuclides of importance in the repository are given in DOE/ET-0029. Calculated doses are directly proportional to the fraction of inventory released; thus, if it were postulated that 10% rather than 1% of the inventory was dispersed, the reported dose would be 10 times higher.

(a) There does not appear to be a direct equivalency between the energy of the meteorite and the nuclear weapon. Claiborne and Gera (1978) conclude that the largest presently deployed missile capable of carrying a 25-megaton bomb would form a 270-m crater; if a 50-megaton bomb were deployed a crater up to 500 m may be formed. Other calculations made for this Statement based on the work of Glasstone (1964) suggest that a bomb on the order of 130-megatons (air blast) would be required to produce a crater 2 km in diameter and 600 m deep.

TABLE 5.5.1. First-Year Whole-Body Dose[(a)] to Maximum Individual--Repository Breach by Meteorite Strike, rem

Time of Event	Salt	Granite	Shale	Basalt
Year of closure				
Spent Fuel	8.3×10^3	2.2×10^4	1.1×10^4	2.2×10^4
Reprocessing Wastes	1.1×10^4	9.2×10^3	6.5×10^3	1.1×10^4
Closure + 1000 years				
Spent Fuel	6.0	1.6×10^1	8.1	1.6×10^1
Reprocessing Wastes	6.2	5.3	3.8	6.2
Closure + 100,000 Years				
Spent Fuel	4.4	1.2×10^1	5.9	1.2×10^1
Reprocessing Wastes	1.1	9.2×10^{-1}	6.6×10^1	1.1
Closure + 1,000,000 Years				
Spent Fuel	2.5	6.6	3.3	6.6
Reprocessing Wastes	7.7×10^{-1}	6.5×10^{-1}	4.7×10^{-2}	7.7×10^{-1}

(a) Doses displayed in Tables 5.5.1 through 5.5.5 reflect relative differences in host rock media only to the extent that different amounts of waste are involved on a per-area basis.

TABLE 5.5.2. 70-Year Whole-Body Dose Commitment to Maximum Individual--Repository Breach by Meteorite Strike, rem

Time of Event	Salt	Granite	Shale	Basalt
Year of closure				
Spent Fuel	3.9×10^6	1.0×10^7	5.1×10^6	1.0×10^7
Reprocessing Wastes	4.7×10^6	4.0×10^6	2.9×10^6	4.7×10^6
Closure + 1000 Years				
Spent Fuel	3.6×10^2	9.5×10^2	4.7×10^2	9.5×10^2
Reprocessing Wastes	3.6×10^2	4.3×10^2	2.2×10^2	3.6×10^2
Closure + 100,000 Years				
Spent Fuel	3.3×10^2	8.9×10^2	4.4×10^2	8.9×10^2
Reprocessing Wastes	3.0×10^1	2.5×10^1	1.8×10^1	3.0×10^1
Closure + 1,000,000 Years				
Spent Fuel	1.7×10^2	4.5×10^2	2.2×10^2	4.5×10^2
Reprocessing Wastes	9.4	7.9	5.8	9.4

The maximum individual, who is 4 km from point of impact, would not survive the initial blast of the meteorite. Regardless, doses in the first year following a release of wastes by a meteorite in the year of closure would amount to 8,000 to 22,000 rem to the whole-body, either of which as an acute dose would prove fatal.

An estimate was made of the number of persons in the reference environment surrounding the repository who could be expected to receive at least 500 rem in the first year following meteorite impact. This was done by calculating the ratio of the atmospheric dispersion coefficients at various points of the compass and the distance from the point of contact. The number of persons so exposed amounted to about 30,000 for the midwest site. If this dose is received in a short time, it would prove fatal to about half of these individuals; thus about 15,000 early radiation-related fatalities would be expected.

Doses to the maximum individual for a breach by meteorite 1000 years after closure range from about 1/3 to 3 times the currently applicable occupational limit and in terms of accidental exposure are not particularly noteworthy. Dose to the maximum individual as a function of time of repository breach decreases slowly after the first thousand years. For a breach at one million years, the dose would vary from about 1% to 100% of applicable occupational dose limits.

Doses to the regional population (2 million persons within 80 km) were calculated and are presented in Table 5.5.3.

TABLE 5.5.3. 70-Year Whole Body-Dose Commitment to the Regional Population--Repository Breach by Meteorite, man-rem

Time of Event	Salt	Granite	Shale	Basalt
Year of closure				
Spent Fuel	6.9×10^7	1.8×10^8	9.1×10^7	1.8×10^8
Reprocessing Wastes	6.2×10^7	5.3×10^7	3.8×10^7	6.2×10^7
Closure + 1000 Years				
Spent Fuel	1.6×10^7	4.2×10^7	2.1×10^7	4.2×10^7
Reprocessing Wastes	6.2×10^6	5.3×10^6	3.8×10^6	6.2×10^6
Closure + 100,000 Years				
Spent Fuel	2.8×10^5	7.4×10^5	3.7×10^5	7.4×10^5
Reprocessing Wastes	7.8×10^4	6.6×10^4	4.8×10^4	7.8×10^5
Closure + 100,000,000 Years				
Spent Fuel	9.4×10^4	2.5×10^5	1.2×10^5	2.5×10^5
Reprocessing Wastes	8.5×10^4	7.0×10^4	5.1×10^4	8.5×10^4

The population dose from a meteorite breach of a single repository in the year of closure would range from 3.8×10^7 to 1.8×10^8 man-rem.[a] About 3.8×10^3 to 1.4×10^5 health effects[b] might be expected from this event. For a breach in the year of closure, the dose to the regional population is about 1 to 10 times the dose received from naturally occurring sources.

As shown in Table 5.5.4, the dose for the second and subsequent generations (70 years per generation) of residents in the regional population is substantially smaller than that for the first generation. The range of doses for the second generation (from 1.1×10^3 to 2.8×10^3 man-rem) may be compared to the dose from naturally occurring sources over the same 70-yr period of 1.4×10^7 man-rem.

TABLE 5.5.4. 70-Year Cumulative Whole-Body Dose to First Five Generations[a] of Regional Population--Repository Breach by Meteorite, man-rem

Spent Fuel Repository	Salt	Granite	Shale	Basalt
Generation				
1	6.9×10^7	1.8×10^8	9.1×10^7	1.8×10^8
2	1.1×10^3	2.8×10^3	1.4×10^3	2.8×10^3
3	2.1×10^2	5.5×10^2	2.7×10^2	5.5×10^2
4	6.3×10^1		- not calculated	
5	1.3×10^1		- not calculated	
Reprocessing Waste Repository				
Generation				
1	6.0×10^7	5.1×10^7	3.6×10^7	6.0×10^7
2	1.2×10^3	1.1×10^3	7.5×10^2	1.2×10^3
3	2.4×10^2	2.1×10^2	1.5×10^2	2.4×10^2
4	5.5×10^1		- not calculated	
5	1.2×10^1		- not calculated	

(a) A generation is taken here to mean 70 years. At the end of that time the population is replaced by an identical population that lives for 70 years.

Within the reference environment (midwest), 150 persons reside within 3.2 km of the repository center, the point of meteor impact. All of these people are presumed to be killed by the blast and thermal effects. A similar meteorite impacting in the metropolitan area of city G in the reference environment (50 to 80 km away) would result in about 25,000 immediate fatalities within a 3.2 km radius. No thought is apparently given by the public to the potential for societal loss from meteorites striking urban areas. Similarly little concern should be had for meteorites striking a waste repository, particularly since calculated consequences are somewhat less for the meteorite case.

(a) Normalizing the 70-yr whole-body dose commitment from breach of a repository by meteorite to the electrical energy produced yields 5.5×10^4 man-rem/GWe-yr for the once-through fuel cycle and 3.6×10^4 man-rem/GWe-yr for the reprocessing cycle.
(b) Using the range of 100 to 800 health effects per million man-rem conversion factor between dose and effect. See Appendix E for details.

Doses to the population of the eastern half of the United States were also calculated and are presented in Table 5.5.5. An assumption is that the prevailing winds in the upper atmosphere will move the radionuclides released during the accident in an eastward direction, which will expose about 160 million persons east of the midwest reference site. The 2 million persons in the reference population are excluded from this calculation. See DOE/ET 0029, Sec. 4.4.3, for additional assumptions used in these calculations. The largest tabulated whole-body dose to the eastern U.S. population of 1.5×10^8 man-rem from meteorite breach of spent fuel repository in the year of closure may be compared with the 1.1×10^9 man-rem this population would receive from naturally occurring radiation sources over the same time period.

TABLE 5.5.5. 70-Year Whole-Body Dose Commitment to Population of Eastern United States--Repository Breach by Meteorite Strike, man-rem

Time of Event	Salt	Granite	Shale	Basalt
Year of closure				
Spent Fuel	5.6×10^7	1.5×10^8	7.4×10^7	1.5×10^8
Reprocessing Wastes	5.2×10^7	4.4×10^7	3.2×10^7	5.2×10^7
Closure + 1000 Years				
Spent Fuel	1.0×10^7	2.7×10^7	1.3×10^7	2.7×10^7
Reprocessing Wastes	3.8×10^6	3.2×10^6	2.3×10^6	3.8×10^6
Closure + 100,000 Years				
Spent Fuel	1.8×10^5	4.8×10^5	2.4×10^5	4.8×10^5
Reprocessing Wastes	4.9×10^4	4.2×10^4	3.0×10^4	4.9×10^4
Closure + 1,000,000 Years				
Spent Fuel	6.3×10^4	1.7×10^5	8.5×10^4	1.7×10^5
Reprocessing Wastes	5.2×10^4	4.4×10^4	3.2×10^4	5.2×10^4

If a meteorite of the size described impacted anywhere in the nation, the area would probably be declared a disaster area regardless of whether or not it impacted over a waste repository. If a waste repository was nearby, monitoring teams could be dispatched to determine the levels of contamination in air, soils and water. Mitigating action would depend on the levels of activity found in various media and the areas involved. Action would range from withholding crops from use and moving dairy and beef animals to less contaminated areas, to removing contaminated soil where necessary and disposing of it under suitable controls.

The probability of a meteorite capable of striking the surface over the repository and producing a crater 2 km in diameter at the surface has been estimated to be 2×10^{-13} per year (Claiborne and Gera 1974). If the "mathematical expectation of societal risk" is taken as probability times consequence, the societal risk of death or serious genetic defect would be from 4×10^{-3} to 3×10^{-2} health effects from the largest dose to the population as presented in Tables 5.5.3 to 5.5.5 over one million years. By way of perspective, in the United States the societal risk of death by lightning is about 120 per year, or about 1×10^{8} deaths per million years (Accident Facts 1974). Thus, in this framework, the societal risk from a meteorite breach of a repository is about 3×10^{-10} that from lightning strikes. Even if the estimate of probability of this meteorite event was in error by a factor of a billion (as might be the case for the probability of a nuclear detonation over the repository), the risk to society remains less than that from lightning and can hardly be considered significant.

5.5.2 Breach of Repository by Fault, Fracture, and Flooding

This scenario is a combination of improbable events: first, a fracture or series of fractures either from the surface or from near an aquifer penetrates to the repository, second, the fractures are connected and permit water to reach the wastes. Two cases are presented, one where a fairly large stream of water penetrates the repository and leaches out radionuclides and then, following an assumed conduit, returns to the surface to form a stream. The second case presumes water reaches the repository and leaches out radionuclides and transports them beyond the boundaries of the host rock; some of the nuclides are then held up by adsorption on soils outside the repository area before slowly working their way to the biosphere. Such scenarios are presented as being independent of host rock properties.

These scenarios involve improbable combinations of events with very low probabilities of occurrence, and in some cases are contrary to the evidence available. For example, faulting of thick salt units does not generally lead to formation of permeable zones, and the plastic behavior of salt tends to heal any opening. Most of the known faults in salt formations confirm this self-healing behavior of salt (Claiborne and Gera 1974). Also, massive salt units generally occur in a geologic environment that contains clays, shales and argillaceous units that again tend to deform plastically. Faults in rock material that yield by brittle fracture (granite, basalt, some carbonates) are more likely to form permeable zones of crushed, broken rock than faults in salt. However, even in brittle rocks a fault zone may, through the grinding and crushing of the material, form a zone of very low to essentially no permeability. That any fault would form a continuously permeable conduit to the repository is doubtful, even if a fault should occur through the repository to the land surface.

In this scenario the repository is assumed to be breached by fracturing either at 1000, 10,000 or 1,000,000 years after repository closure. Water in the form of a stream of 2.8 m^3/sec[(a)] (100 cfs) invades the repository, flows among the wastes and enters the reference environment in the R river about 10 km from the repository center. The stream is assumed to be in contact with the wastes for one year. (This case simulates the subsequent

(a) Several comments were received on the draft Statement that such a large flow of water was unreasonable. However, the scenario is not all that unreasonable, at least in the long term. One can envision stream displacement as a result of ice dams, glaciation, or land slides to where the scenario becomes plausible at least to the extent of entry of water. Return of water to the biosphere is harder to imagine.

sealing of the breach line by further earth movement, healing because of the nature of the host rock or because of plugging of the water path by silt carried by the stream.)

Several studies have been performed to estimate the leach rate of waste by water. Two important factors affecting leach rate of a waste material are the waste form (chemical nature) and the temperature of the solid-liquid interaction zone. Data reported by Ross (1978), under repository conditions much more severe than would exist a thousand years after closure, indicate leach rates ranging from 10^{-8} to 10^{-5} g/cm^2-day for reactions between aqueous solutions and waste glasses in a devitrified and fractured state. Other studies by McCarthy et al. (1978), with conditions of 300°C and 300 atmospheres, have suggested changes in waste form properties which might lead to higher leach rates for some radionuclides in borosilicate glass. The same processes also caused recombination of some of the radionuclides with the immediate environment to a more stable form with a lower leach rate. Other studies in field situations at lower temperatures and pressure with the ground saturated with water have shown rates as low or lower than 10^{-10} gm/cm^2-day for radionuclides in nepheline syenite glass (Merritt 1976). The leach rates used in consequence analyses, Table 5.5.6, are considered highly conservative in view of these studies and the likely temperature of the water contacting the waste.

TABLE 5.5.6. Estimated Leach Rates for Various Forms of Radioactive Wastes Used in Consequence Analyses.

Waste Form and Assumed Geometry	Leach Rate gm/cm2-day	Number of Canisters Contacted
High-level waste glass (assumed to be devitrified and fractured, and without any protection from the canister--1-cm cubes)	1×10^{-4} for first 10 days 1×10^{-5} thereafter	210
Spent fuel (1-cm-dia spheres)[a]	1×10^{-5}	1230 PWR, 1320 BWR[b]
Fuel residue	1×10^{-5}	30
Other TRU wastes	1×10^{-4}	480, 560

(a) The fuel pellet simulating a combination of PWR and BWR fuels is taken to be a cylinder 1.16 cm in diameter by 1.16 cm long. Since the spent fuel dose calculations were made, the determination has been made that spent fuel may be fragmented following irradiation and that the area subject to leaching may be about 5 times that used in the original calculations (Pasupathi 1978). This factor has been applied to doses in this section.

(b) Subsequent to the calculations made for this Statement on the basis of 1230 PWR and 1320 BWR canisters (816-MTHM) contacted by water and subjected to leaching, the contents of the repositories in the various media were changed. The amounts of spent fuel contacted by water following a 12-m-wide fracture along the diagonal of a repository were estimated to be: salt; 340 MTHM, Granite; 870 MTHM, shale; 390 MTHM and basalt 810 MTHM (DOE/ET-0029). For all practical purposes the doses that follow would apply to the breach of granite and basalt repositories. Doses should be multiplied by a factor of 0.4 to obtain doses reflecting a breach in a salt repository and by a factor of 0.5 for a shale repository.

For dose calculations for spent fuel and vitrified high-level waste (the major contributor to dose from reprocessing wastes), doses may be calculated for other leach rates by multiplying the tabulated dose by the ratio of the assumed leach rates to the listed leach rate.

Seventy-year whole-body dose commitments have been calculated for the maximum individual using the data of Tables 4.4.3 and 9.3.34 in DOE/ET-0029, the methods described in Appendix D and the following assumptions. For cases in other than a salt repository, aquatic food is taken from, and recreational activities occur near, the 2.8 m^3/sec stream of water from the repository (this assumption is perhaps overly simplistic since the stream flows for only one year and little time is available for an aquatic ecosystem to be established). Drinking water is taken from the river downstream from the point of contamination entry (the majority of the regional population resides down stream from the repository and the presumed point of entry of the stream). Contaminants in farm products and ground contamination doses were determined based on irrigation of land with water from the river. In the case of a repository in salt it was concluded that the 2.8 m^3/sec effluent stream would be so laden with salt that no fresh-water biota would be present and that the maximum individual would derive his aquatic food from the river as opposed to the small stream.

Doses to the maximum individual are presented in Table 5.5.7. Population doses were also calculated on the basis of contamination of water in the R river. Seventy-year dose commitments to the maximum individual and the regional population were calculated for 1000, 100,000 and 1,000,000 years after closure of the repository.[(a)] Doses to the regional population are presented in Table 5.5.8. Doses to other regions and for the breach in the year of repository closure may be found in DOE/ET-0029.

The range of population dose for the flooding and faulting event 1000 years after closure amounted to 8.8 x 10^4 to 1.7 x 10^5 for spent fuel and reprocessing wastes, respectively. Using the range of 100 to 800 health effects per million man-rem, the calculated total number of health effects attributable to this event, if it occurred as postulated, would be 9 to 140 depending on fuel cycle.

The probability of a fault intersecting the repository in a typical bedded salt basin such as the Delaware Basin has been estimated by Claiborne and Gera (1974) to be

TABLE 5.5.7. 70-Year Whole-Body Dose Commitment to Maximum Individual--Repository Breach by Faulting and Flooding, rem

Time of Event	Salt Media	Non-salt Media
Closure + 1,000 Years		
Spent Fuel	3.0 x 10^{-1}	9.7
Reprocessing Waste	5.5 x 10^{-1}	1.5 x 10^1
Closure + 100,000 Years		
Spent Fuel	3.7 x 10^{-1}	8.6
Reprocessing waste	6.7 x 10^{-2}	1.5
Closure + 1,000,000 Years		
Spent Fuel	1.8 x 10^{-1}	4.3
Reprocessing waste	2.2 x 10^{-2}	4.5 x 10^{-1}

(a) Calculations were presented in the Draft DOE/EIS 0046-D for a stream breach in 2050. In deference to comments on the unreasonableness of this event, it is not presented here; detection would be almost certain and mitigation of affects possible. At 1000 years after closure the unrecognized contaminated stream does not seem unreasonable.

TABLE 5.5.8. 70-Year Whole-Body Dose Commitment to the Regional Population--Repository Breach by Faulting and Flooding

Time of Event	Man-rem
Closure + 1,000 Years	
Spent Fuel	8.8×10^4
Reprocessing waste	1.7×10^5
Closure + 1,000,000 Years	
Spent Fuel	1.4×10^{5}(*)
Reprocessing waste	2.8×10^4
Closure + 1,000,000 Years	
Spent Fuel	7.1×10^4
Reprocessing waste	1.0×10^4

(*) The increase in dose between breaches at +1,000 and +100,000 years is due principally to the ingestion of ^{226}Ra from the decay chain of ^{238}Pu.

4×10^{-11}/yr. The frequency that a high pressure aquifer exists with canister and surface access is 0.005 (DOE/ET-0028, Sec. 7.4.9). A total probability for release to the biosphere is 2×10^{-13} per year.

Using the probability estimate of 2×10^{-13}/yr and the largest number of health effects calculated, 140 (Table 5.5.8), the mathematical expectation of societal risk would be at most 3×10^{-11}/yr or 3×10^{-7} health effects over 10,000 yr.[(a)]

The population dose to the regional population from naturally occurring sources would amount to about 1.4×10^7 man-rem over the same time period. Even in the maximum case, that of 1.7×10^5 man-rem associated with release of radioactive material from nonsalt repositories, the doses are on the order of 1% of that from naturally occurring sources.

One of the potential long-term effects of release of radionuclides to the river would include the movement of these radionuclides to the ocean, where accumulation in mollusks may occur resulting in another pathway to human exposure. It was assumed that the following dilution factors[(b)] were appropriate for concentrations of elements in an estuary; e.g., concentration of cobalt nuclides in estuary water would be 0.01 of their concenrations in the river.

(a) EPA commented that the calculation of probability was incorrect (see EPA ltr. comment #86; Vol. 3 App C. p 34). The EPA estimate of the probability of a faulting and water intrusion event was 4×10^{-7} over a 10,000-year period compared to 2×10^{-9} (2×10^{-13}/yr $\times 1 \times 10^4$ yr) used in this Statement. EPA concluded that once a fault intersected the repository that the probability of water intrusion in the long term would likely be one. DOE believes the EPA argument has merit, however using the EPA figures increases the societal risk to only 6×10^{-5} over the 10,000 year period, which is still an insignificant societal risk.

(b) Dilution factors are highly dependent on the specific river system and estuary of interest. The dilution factors presented here were developed for movement of radionuclides from reactor effluent water at the Hanford Project in Eastern Washington via the Columbia River to Willapa Bay, Washington, where oysters are harvested.

Element	Dilution Factor	Element	Dilution Factor
H	2	Cs	100
C	2	Sm	100
Co	100	Eu	100
Ni	100	U	100
Sr	100	Np	100
Nb, Zr	100	Pu	100
Sb	2	Am	100
Sn	2	Cm	100
I	2		

Saltwater bioaccumulation factors were used to estimate the concentration of radionuclides in the edible portion of marine foods (Soldat, Robinson and Baker 1974). The 70-yr dose to the maximum individual from ingestion of mollusks (at a rate of 10 kg/yr) for repository breaches at 1,000, 100,000 and 1,000,000 years after repository closure were calculated. The largest of these, 7.2×10^{-2} rem to the whole-body, would add about 1% to the dose the individual would have received from naturally occurring sources for the same period and would not add significantly to the maximum individual's 70-year dose commitment.

The second scenario developed for the repository fracture and flooding assumes that radionuclides are leached from the waste and carried beyond the boundaries of the host rock and are then transported via moving (100 m/yr) ground water through the ground before entering the biosphere (the R river).

In this scenario a migration path length of 10 km was investigated, using sorption equibrium constants (Kd values) measured or estimated under conditions at the Hanford Site, Richland, Washington. While these parameters are believed to be representative of average conditions to be expected at candidate sites, all factors could vary by several orders of magnitude.

Based on inventories of radionuclides in repositories and the models and dose calculation methods according to Lester et al. (1975) and Burkholder et al. (1975), doses were calculated for the maximum individual.[a] Total body doses are presented in Table 5.5.9 as a function of time since disposal and for leach rates ranging from 0.1% to 0.01% of inventory per year.[b]

The doses given in Table 5.5.9 were calculated to result from leaching of all wastes from a 50,000 MTHM example repository in salt. These doses would be about 2.5 times higher for the repositories in granite or basalt and about 1.3 times higher should the event occur in a shale repository due to larger amounts of waste contained in those repositories. In

(a) A computer model called GETOUT for hydrologic transport (Lester et al. 1978) was used in conjunction with a dose to biota model (Burkholder et al. 1975) as adjusted for parameters developed for the midwest reference environment.

(b) Several commenters on the draft concluded that total release of inventory in one year as presented in the draft Statement was out of the question. As a consequence the 100% removal per year case is omitted. The leach rates of 1×10^{-5} g/cm^2-day used in the fracturing and stream flooding scenario amounts to about 1% of inventory removed per year, using assumptions that maximize the area available to contact water.

TABLE 5.5.9. 70-Yr[*] Accumulated Whole-Body Dose to Maximum Individual for Various Leach Rates and Times of Repository Breach by Fracturing and Ground-Water Intrusion (repository in salt--50,000 MTHM), rem

	Spent Fuel				Reprocessing Wastes			
	Breach 1000 years after closure, Leach Rate (yr^{-1})		Breach 100,000 years after closure, Leach Rate (yr^{-1})		Breach 1000 years after closure, Leach Rate (yr^{-1})		Breach 100,000 years after closure, Leach Rate (yr^{-1})	
Years Since Disposal	0.1%	0.01%	0.1%	0.01%	0.1%	0.01%	0.1%	0.01%
1.0×10^3	5×10^{-2}	5×10^{-3}			5×10^{-2}	5×10^{-3}		
2.0×10^3	1	1×10^{-1}			4×10^{-1}	4×1^{-2}		
1.0×10^4	5×10^{-1}	8×10^{-2}			5×10^{-1}	5×10^{-2}		
3.4×10^4	5×10^{-2}				5×10^{-2}			
1.1×10^5			2×10^{-2}	4×10^{-3}			3×10^{-2}	3×10^{-2}

(*) The computer program for this scenario used 50 rather than 70 years for exposure purposes. The values tabulated were adjusted upward for an additional 20-year exposure.

each case the host rock was assumed to be surrounded by a common soil-rock medium for which absorption rates would be the same.

The largest dose tabulated was 1 rem over 70 years if the event should occur. This is about one-seventh of the dose the individual would have received from naturally occurring sources and is believed to be of no consequence. The probability of this event occurring over a 10,000 year period is estimated to be in the neighborhood of 4×10^{-7} to 2×10^{-9}.[a]

Over a time span of 100,000 years a peak dose occurs that is essentially independent of leach rate or time of repository breach. The dose is due principally to ^{226}Ra, decay product of ^{238}U (which has an extremely long half-life).[b] At 1.4 million years after disposal the 70-yr dose to the maximum individual amounted to about 70 rem. This long-term radiological risk would not be significantly different from that of a natural ore body of similar content.

Doses to the regional population were not calculated directly for this scenario; rather, an estimate was made using a ratio of the per capita population whole-body dose and the whole-body dose to the maximum individual in the previously presented 2.8 m^3/sec stream scenario. The ratio obtained was 1/5 and thus the per capita population dose was approximately one-fifth of the maximum individual dose. A whole-body dose to the regional population from ground-water contamination from breach of a 50,000 MTHM repository was estimated by multiplying the per capita dose by 2 million, the size of the regional population. Taking the largest maximum individual dose of 1 rem over 70 years to the whole body and using

(a) Probability of faulting over a 10,000 yr period of 4×10^{-7} was taken from EPA comment #113 on the draft to this statement. The probability of 2×10^{-9} over 10,000 years was developed from Claiborne and Gera (1974).
(b) About 10% of ^{226}Ra is a result of decay of ^{238}Pu produced in the reactor. About 90% of the ^{226}Ra is from unaltered ^{238}U in the fuel. After long periods of time, the principal source of potential dose to the public is the uranium from which the reactor fuel was made.

this conversion, a regional population dose of about 2×10^5 man-rem is obtained.[a] By comparison the dose to this population from naturally occurring sources over the same period would be about 1.4×10^7 man-rem.

Unlike some of the other scenarios the contamination in this event could be expected to reach the environment continuously over a long period of time. For example, the 70-year dose to the maximum individual decreased from 1 rem to 0.5 rem between 2000 and 10,000 years after disposal (a factor of 2 would be lost in the imprecision of the estimate). The total dose to replicate regional populations over 10,000 years would be on the order of 3×10^7 man-rem (143-70 yr generations). The total regional population dose for this same period from naturally occurring sources would be about 2×10^9 man-rem. As noted earlier the probability of this event occurring is estimated to be between 4×10^{-11} and 2×10^{-13}/yr. The probability that it would occur sometime within a 10,000-yr period would be on the order of 4×10^{-7} to 2×10^{-9}. The mathematical expectation of societal risk would be less than one fatality over 10,000 years.

5.5.3 Faulting and Ground-water Intrusion to a Domestic Well

In this scenario a fault intersects a repository (non-salt) and water from an aquifer beneath the repository flows in small quantity through the repository to an overlying aquifer that is tapped by a domestic well. The domestic well is postulated to be located about 3 km down gradient from the fault and is capable of producing about 20 liters of potable water per minute.

In order to estimate the maximum consequences that might occur from the interaction with the buried waste, the assumption is made that all water flowing through the fault enters the domestic well. This suggests that the upper aquifer is of low permeability. Most domestic production wells are not drilled in aquifers of low permeability. Thus, for more usual permeabilities encountered, a much smaller fraction of the waste nuclides would arrive at the well. The water travel time from the fault in the repository to the domestic well would vary from 1000 to 2500 years depending on the streamline the water followed between the source and the well, while transport times for radionuclides could vary from a thousand to millions of years depending on the nature of the radionuclides and the sorption characteristics of the medium through which the water was flowing.

Doses were calculated from the rupture and leaching of 1320 BWR fuel assemblies and 1230 PWR fuel assemblies for the spent fuel repository; and 210 high-level waste canisters, 30 RH-TRU waste canisters and 480 barrels of RH-TRU waste for the reprocessing waste repository. All of the stated radioactive content is leached out over a 10,000-yr period.

The maximum 70-yr accumulated whole-body doses to the maximum individual from specific long-lived waste radionuclides that may be of interest and the time after connection with

(a) In reviewing the Draft EIS, EPA criticized this approach to population dose. At best, the method is a crude approximation of the population dose; but this approximation was made in lieu of reprogramming an existing dose code solely for this purpose. In any event the population dose could not exceed the dose of the maximum individual times the regional population (2×10^6 man-rem) and would likely be substantially less. (As in the previous scenario of a stream through repository, most of the population resides down stream from the entry of contaminated water.)

the repository that the dose would occur are as follows. Assuming that ^{129}I removed from dissolver off-gas is sent to the repository and is leached at roughly the same rate as from spent fuel, the doses are essentially the same for either fuel cycle option.

Radionuclide	Dose, rem	Time, yr
^{14}C	90	1×10^4
^{99}Tc	22	4×10^3
^{129}I	990 (to the thyroid)	1×10^4
^{135}Cs	0.2	1×10^6
^{237}Np	440	1×10^6

The probability of the event is estimated to be on the order of 4×10^{-7} to 2×10^{-9} over a 10,000-yr period.[a]

Because of the extremely small probability of occurrence, and because of the very limited number of individuals that could be contaminated by such a well, the societal risk is believed to be insignificant.

5.5.4 Repository Breach by Drilling

In this scenario, about 1000 years after repository closure an individual (or group) drills 600 m into a waste repository in search of a mineral resource or for geologic study itself. Repository markers are no longer evident, are misunderstood, or are ignored. These individuals, while having the technology to drill to repository depth, do not possess or do not apply the knowledge and apparatus to assay material brought up in the drilling process and to discover its radioactive properties.[b]

Because a probability for exploratory drilling could not be determined, an overall probability was not assigned to this event. In qualitative terms, someone could be exploring for potash, oil, etc.[c] in the area of a repository in salt based on the same exploration principles that established the presence of the formation in the first place. In other formations such as granite, shale and basalt, associations with any particular resources are not as strong as in the case of salt. The probability that drilling will occur somewhere on the repository site is highly uncertain. _If_ drilling occurs on the property, the

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(a) Probability of faulting over a 10,000-yr period of 4×10^{-7} was taken from EPA comment #113 on the draft to this statement. The probability of 2×10^{-9} over 10,000 years was developed from Claiborne and Gera (1974).

(b) The drill crew may not be aware of radioactive material in the drilling mud as it is brought up; however, once samples are sent to their assay laboratory, the drillers would soon know of the radioactive nature of their exploratory effort. If the assay were crude they might conclude, in the case of drilling through a spent fuel element, that they had struck uranium, but very little sophistication in assay would be required to recognize that the radiation spectrum was not at all like that-of natural uranium. The radiation characteristics of material brought up after passing through a solidified high-level waste canister would resemble natural ores even less.

(c) Because of the frequent occurrence of salt deposits at depths much shallower than 600 m the explorer would not likely be drilling to 600 m in search of salt.

probability that the drill (0.5 m in diameter) will strike a waste canister is 0.005 per drilling event, because of the relative cross sectional areas involved.

For dose calculations it is assumed that during drilling one-fourth of the waste in one canister is circulated to the surface with the drilling mud, and the radioactive material is uniformly distributed over 0.5 ha in the top 5 cm of the surface soil.

Table 5.5.10 lists the expected releases to air from contaminated surface soil. These values are based upon 1) a resuspension factor of 0.011/yr 2) the assumption that one-fourth of the radioactive material in the top 5 cm is available for resuspension and 3) that 0.10 of the material resuspended is respirable. The maximum individual is exposed, on the average, to the contaminated soil for 12 hr/day. Based on the releases given in Table 5.5.10 and methods of dose calculations presented in Appendix D, first-year doses and 70-yr doses to the maximum individual who will reside and grow crops for his consumption on the contaminated land were calculated. The first-year whole-body doses amounted to 13 rem for drilling through a spent fuel canister and 19 rem for drilling through a HLW waste canister. The 70-yr whole body doses were 9.4×10^2 and 1.4×10^3 rem, respectively.

TABLE 5.5.10. Respirable Radionuclides Released to the Atmosphere from Salt Repository Breach by Drilling 1000 Yrs After Repository Closure, Ci

Radionuclide	Spent Fuel	HLW
^{14}C(*)	1.7×10^{-4}	1.1×10^{-6}
^{126}Sb	1.2×10^{-5}	1.1×10^{-4}
^{126}Sn	1.2×10^{-5}	1.1×10^{-4}
^{129}I(*)	8.6×10^{-6}	7.4×10^{-7}
^{239}Np	3.4×10^{-4}	9.0×10^{-3}
^{240}Pu	1.1×10^{-2}	4.4×10^{-4}
^{241}Am	2.2×10^{-2}	3.2×10^{-2}
^{239}Pu	7.3×10^{-2}	6.3×10^{-4}

(*) The bulk of the C and I is volatized during dissolution of the spent fuel and stored in separate containers and locations different than those used for HLW in the repository. For these two nuclides, 100% of the material resuspended is assumed respirable.

The predominant mode of exposure is direct radiation[(a)] from contaminated soil and as a consequence, dose to the various organs is substantially the same the first year. During the 70-yr dose period the dose via the ingestion pathway increases substantially, particularly in terms of dose to bone. The 70 year accumulated doses as calculated might result in a small increase in risk of life shortening, contracting lukemia, etc.

(a) ^{241}Am is the principal contributor to the direct radiation dose. The dose from breach of a HLW canister was reported in the draft Statement, and in supporting documents, as about 100 times higher than here because an incorrect ^{241}Am inventory was used.

If the 0.5 ha of contaminated land were occupied by a housing project soon after the drilling incident with about 0.1 ha per lot, five families (probably about 25 individuals) might be exposed to the same extent as the maximum individual.

Seventy-year accumulated doses calculated for the regional population amounted to 1.1×10^2 man-rem in the case of spent fuel and 1.6×10^2 man-rem in the case of reprocessing wastes. All of the doses to the regional population (whose exposure would result principally from resuspension and air transport of radionuclides) are substantially less than those which would be received from naturally occurring radioactive sources (1.4×10^7 man-rem over the same period).

In the case of a repository in salt, the land (0.5 ha) would likely be contaminated with salt brought up with the drilling mud. As developed in more detail in DOE/ET-0029 the resulting ground contaminated by salt would not be well tolerated by ordinary crops.

Breach of a waste canister by exploratory drilling, if it occurred, could result in a small increase in risk of adverse health effects occurring among about two dozen people in the immediate area.

If exploratory drilling that reached the repository level were abandoned (whether a canister had been penetrated or not) it could provide a means of entry of water into the repository. It is believed that the bore hole would not remain open for long but if it did and significant quantities of water were to flow in and out the consequences would not reasonably exceed those described previously for faulting and flooding of a repository.

The key to mitigating action associated with a drilling accident is the discovery that radioactive material had been encountered. As stated, that knowledge would probably come from assay of the drill core or samples of the drilling mud. If the driller is aware that a drill has brought waste to the surface, standard decontamination methods could be used to recover the contaminants, dispose of them under suitable controls, and preclude essentially all of the previously mentioned radiological consequences.

5.5.5 Solution Mining

In this scenario a 47,000 MTHM example geologic repository in domed salt[a] is breached by solution mining 1000 years after the repository is closed. Although this accident is typified by solution mining for salt recovery, solution mining is also used for extraction of other resources and for construction of underground storage cavities. This accidental breach of a repository is believed to be conceivable only for an industrialized society having technological capabilities substantially as exist today.

Basically, solution mining in domed salt involves drilling a well to the desired level and inserting a double-walled pipe so that water can be forced down the outer pipe into the salt, where it dissolves the salt into a brine and forces the brine back up through the center pipe (Kaufmann 1960). The life of such solution wells varies markedly, some failing in

(a) Solution mining of stratified salt is believed less likely than in dome salt because of less evidence suggesting the presence of salt.

a few years. For purposes of this accident analysis the well(s) could operate for 50 years before being abandoned because of failure caused by cave-in and crushing and plugging of piping with debris.

This accidental repository intrusion, as in the case of the drilling accident, is based on the assumption that repository markers are either no longer evident, are misunderstood or ignored. Salt deposits are relatively plentiful and drilling to 600 m for salt seems highly unrealistic. No probabilities could be assigned to this event; it is presented only as a hypothetical "what if" accident.

Ordinarily, once the brines are brought to the surface they are analyzed to determine the kinds and amounts of impurities such as calcium sulfate, calcium-magnesium carbonate, sulfides, etc., which would govern further processing to purify the salt. If radioactive waste is placed in repositories in salt formations, salt used for human consumption could be checked by radioanalysis as well (an institutionally administered precaution). Calculations suggest that radioactivity would be determinable with off-the-shelf gamma-ray spectrometer apparatus on samples of a few hundred grams at concentrations of waste in salt existing after a few days of mining operation and with certainty by one month of mining operation.

Assumptions of the scenario are that, although the salt stratum of the reference site is about 80 m thick, the salt removed is principally that from backfill, ceiling, pillars and floor where radioactive waste has been placed. In mining the repository about 33 million tons of salt would have been removed for waste placement. This represents about one-fourth of the total salt volume in the mined area (in the scenario, the repository has been backfilled completely with salt; actually backfill of about 60% is presently planned). The total salt postulated to be solution mined over 50 years is then about 130 million tons.[a] This represents about 10% of the total salt contained in the salt stratum bounded by the repository area. If an equal amount of salt is mined in each of 50 years, the annual production would be about 2.6 million tons. In 1957 about 24 million tons of salt were produced in the United States (Kaufmann 1960). Such a solution mining operation for salt would exceed the size of those presently in operation in the United States; a very large operation in the United States produces about 0.4 million tons annually and in Europe a very large operation may produce on the order of 1 million tons of salt annually.

Given that 100 parts of water (at 20 to 100 C) by weight can dissolve 36 to 39 parts of salt, then over a 50-yr period a stream flow of 210 /sec is required to dissolve that much

(a) Although it is believed that radioanalysis of salt would result in termination of the operation soon after start-up, the scenario is developed based on removal of the repository salt over a 50-yr period. Amounts of wastes and salt brought to the surface over shorter periods of time are pro-rated based on water contact with all wastes by the end of 50 years. Consequences are based on the assumption that the presence of radioactivity goes undetected for one year.

salt. If an adequate source of water is available, nine wells each operating at about 23 /sec would be sufficient.

The actual solution chemistry of leached radionuclides moving into the salt brine is not known. An assumption of the scenario is that radionuclides leached from spent fuel mix completely with the salt brine and are carried to the surface. Although it may take 1/2 to 1-1/2 years to bring a brine well to production, in the scenario, the brine well produces immediately and continuously for 50 years, at the end of which the entire quantity of salt surrounding the waste would have been mined out. Water flow would follow a course of least resistance and would follow the previously mined cavern boundaries where possible; this maximizes the consequences.

Details of the calculations for leaching of radionuclides in spent fuel and in reprocessing waste with the disposed salt may be found in Sections 4.4.and 9.3, respectively, of DOE/ET-0029.

If 3% of the 2.4 million metric tons of salt mined per year is used for human consumption, then about 72,000 metric tons would be used for that purpose. If a person consumed 1800 g/yr then 72,000 metric tons of salt would provide for about 40 million persons. For purposes of this analysis the exposed population consists of 40 million persons.

Although daily monitoring controls on the salt would bring attention to the presence of contaminated salt, the possible failure of such monitoring was recognized. The producers' quality assurance laboratory may not recognize the failure for a week. That failing, it might take as much as a year before a consumer discovered the contamination. On this pessimistic series of circumstances the conclusion was that a reasonable upper bound on waste entering the food trade would be that in salt produced in one year. Therefore, the consequences of this accident in terms of radiation dose to an exposed population of 40 million persons from ingestion of contaminated salt for one year were calculated. The quantities of radionuclides which contributed significantly to whole-body dose and the doses are listed in Table 5.5.11.

TABLE 5.5.11. Amounts of Radionuclides (Ci) and 70-Year Whole-Body Dose (in rems to an individual) Resulting from Ingestion of 1800 g of Contaminated Salt 1000 years after Repository Closure

	Curies		rems	
	Spent Fuel Repository	Reprocessing Waste Repository	Spent Fuel Repository	Reprocessing Waste Repository
^{239}Pu	1.5×10^{-6}	2.2×10^{-8}	3.6×10^{-2}	5.5×10^{-4}
^{240}Pu	2.2×10^{-6}	1.2×10^{-7}	5.3×10^{-2}	3.1×10^{-3}
^{241}Am	4.5×10^{-6}	1.3×10^{-6}	3.0×10^{-1}	8.6×10^{-2}
^{243}Am	6.6×10^{-8}	2.4×10^{-7}	4.0×10	1.5×10
Total			3.9×10^{-1}	1.0×10^{-1}

The 70-yr whole-body dose commitment to the exposed population of 40 million persons would amount to 1.6×10^7 man-rem for such an event occurring in a spent fuel repository and to 4.0×10^6 man-rem from a similar event at a repository for reprocessing wastes. These dose commitments are less than one-tenth of the 2.8 x 10^8 man-rem that the exposed population would receive over the same time period from naturally occurring sources. The relatively low population doses that might result if the event occurred indicates that the solution mining event would not constitute a significant societal risk.[a]

(a) Other assessments of a solution mining event have been made in which different assumptions of repository size and amount of radionuclides reaching culinary salt were made. In particular the leaching was limited by the solubility of the uranium content of the waste. The contaminated salt was calculated to be distributed among 15 million persons. The 70-year dose to an individual for this event in a spent fuel repository amounted to 2.3 rem. This dose is about a factor of 6 higher than in the above analysis, but would also result in population doses less than those from naturally occurring sources.

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5.6 COST OF GEOLOGIC DISPOSAL

Constant dollar[a] costs have been estimated for isolating both spent fuel and fuel reprocessing wastes in salt, granite, shale, and basalt formations. The costs include all construction, operating, and decommissioning costs. The costs of federal programs for repository research and development have not been included in the costs stated here, but are included in the systems cost estimates in Chapter 7. The cost estimates are stated in terms of constant 1978 dollars.

Results of this analysis show that for spent fuel repositories of constant size (800 ha), construction costs including mining and backfilling range from $1 billion for bedded salt media to $3 billion for basalt media. Total operating costs vary from $590 million for a repository in salt to $2.4 billion for one in basalt. However, since the allowable waste emplacement density in basalt is about 2.5 times greater than that in salt, unit costs for disposal in basalt are only about 70% higher than for disposal in salt. Costs of disposal in shale are similar to those in salt and costs of disposal in granite are similar to those in basalt. Cost estimates for reprocessing-waste repositories follow a similar pattern.

5.6.1 Construction Costs

The repository construction cost estimates include owner's costs as well as facility construction. Owner's costs include land acquisition, startup costs, owner's staff costs and other costs incurred by the owner--in this case the Federal government or its contractor--during construction. Facility construction costs are defined here to include the costs of all labor, equipment (including waste transport and emplacement equipment), buildings and structures, site improvements, shaft, corridor and room mining, backfilling, and architect/engineer services. Interest during construction is taken into account by discounting prestartup construction costs at 7% per year (constant dollar rate which excludes inflation). Construction cost estimates were generally based on designs prepared by the Office of Waste Isolation (OWI) in documents Y-OWI/TM-36, Vol. 1-23. These designs have been revised somewhat to reflect more efficient shaft design, construction and usage, revised mining schedules, increased surface storage of mined rock, and more workable surface handling facilities (see Vol. 4, Chapter 7 of DOE/ET-0028 (DOE 1979) or Section 5.3 of this Statement for repository descriptions). Construction costs are derived by estimating requirements for major equipment, buildings and structures, site improvements, and construction labor. These direct cost estimates are then factored to generate other direct costs, architect/engineer costs, and owner's costs.

The construction cost estimates, including a contingency factor, have an estimated accuracy range of $\pm$20%. This accuracy range reflects the uncertainties that are likely to be encountered during design and construction, but which are difficult or impossible to

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(a) The term constant dollars means that the dollar value of the estimates in all future time periods is the same as the value of the dollar in the reference year (1978 in this statement); i.e., the effects of inflation are removed.

identify at this time, such as siting and engineering scope requirements necessary to provide a fully functional facility. Also included in the estimates are the possible variances of the assumed rock densities used in the development of mining costs. The contingency factors are such that, within the stated accuracy range, there is an approximately equal likelihood of the indicated cost overrun or underrun. The construction cost estimation methodology is explained in more detail in DOE/ET-0028, Vol. 1, Section 3.8.

Construction costs for repositories in different media are based on a fixed repository area of 800 ha (2000 acres). However, since waste emplacement density is a function of the thermal characteristics of each type of media, actual waste quantities emplaced differ for each 800 ha repository. Table 5.6.1 shows equivalent waste quantities emplaced, the resultant mining requirements and the construction costs. Operating costs and unit costs are also given in this table to facilitate comparisons of cost relationships. These costs are discussed in subsequent sections.

Since mining costs account for 30% to 50% of total construction costs, the total construction costs vary significantly between geologic media. However, emplacement capacity increases for media with higher mining costs (see Section 5.3) and the relative unit cost differences between geologic media are smaller than the relative construction cost differences. For example, construction costs for an 800-ha repository in basalt are about three times those of an 800-ha repository in salt for the once-through cycle, but the cost per

TABLE 5.6.1. Cost Estimates for 800-hectare Geologic Repositories

Waste Type	Geologic Media	Mined Quantity 10^6 MT	Equivalent MTHM of Waste Emplaced		Construction Cost[(a)] Millions of $	Total Operating Cost[(b)] Millions of $	Unit Cost[(c)] $/kg HM[(e)]
Spent Fuel	Salt	30	51,000		1,000	590	52
	Granite	77	121,600		2,600	2,350	78
	Shale	35	64,500		1,300	810	57
	Basalt	90	121,600		3,100	2,390	87
			HLW[(d)]	TRU[(d)]			
Reprocessing Cycle Wastes	Salt	35	62,000	100,000	1,200	1,210	47
	Granite	53	69,000	108,000	2,000	1,940	77
	Shale	30	30,500	56,000	1,300	830	73
	Basalt	59	56,000	92,000	2,300	1,740	93

(a) Includes mining, backfilling and shaft sealing costs. Backfilling and shaft sealing costs are approximately 10% of total construction costs. Uncertainties in construction cost estimates are about ±20%.
(b) Uncertainties in operating cost estimates are about ±25%.
(c) Includes decommissioning costs. Uncertainties in unit cost estimates are about ±50%.
(d) The metric ton of heavy metal (MTHM) equivalent of high-level waste stored at the initial repository is less than the MTHM equivalent of TRU wastes since the high-level waste must be cooled 5 years before it can be sent to the repository and deliveries to the repository lag behind TRU waste deliveries.
(e) Costs may be expressed in $/GW-yr by multiplying by 38,000 KgHM/GW-yr.

kilogram of disposal in a basalt repository is only 67% higher. These unit cost relationships may change somewhat for repositories of optimized size at specific sites.

Construction costs for repositories in granite and basalt are much higher than for repositories in salt and shale, mainly because of mining cost differences. These differences arise because of different mined quantities, as noted previously, and because of higher unit mining costs reflecting the greater difficulty in hard-rock mining.

5.6.2 Operating Costs

Operating costs include the costs of direct labor, monitoring and safety, materials, utilities, maintenance, administrative and other overhead, hole drilling and/or trenching and retrievability sleeve placement. The materials category includes all overpacks, sleeves, and plugs used in the repository. Waste packaging or encapsulation costs were considered to be a predisposal cost and can be found in Section 4.9. Costs of the waste canisters are included in the encapsulation costs in the case of spent fuel or in the waste treatment and packaging costs in the case of reprocessing cycle wastes.

Labor, utilities, and maintenance requirements are based on estimates given in Y/OWI/TM-36, Vol. 10, 12, 14 and 16. Materials requirements, wage rates, and utility costs are based on annual receipts and price data described in DOE/ET-0028, Vol. 1, Section 3.8.2. Unit hole drilling, trenching, and sleeve placement costs were derived by the architect/engineer making the construction cost estimates and are detailed in DOE/ET-0028, Vol. 4, Sections 7.4.10.2 and 7.5.10.2. The allowances for maintenance, overhead, and miscellaneous costs have been derived by factoring either capital or direct labor costs. After inclusion of a 25% contingency factor the operating cost estimates have an estimated uncertainty of approximately ±25%.

Total operating costs for the waste repositories are shown in the sixth column of Table 5.6.1. These figures represent the cumulative operating costs during the repository waste receiving periods. Cumulative operating cost differences between repositories are principally due to differences in amount of waste emplaced. The granite and basalt repositories generally have significantly higher cumulative operating costs than do repositories in salt and shale because of their greater capacity and longer operating lifetimes. Another significant factor in operating cost differences in spent fuel repositories is the higher cost of hole drilling in granite and basalt for canister placement.

5.6.3 Decommissioning Costs

Decommissioning costs are defined here to include decommissioning of the surface facilities and sealing of the repository shafts. Based on decommissioning cost estimates for other fuel cycle facilities, the decommissioning cost of the repository surface facilities is estimated at 10% of the construction cost of these facilities. Shaft sealing costs are estimated to be approximately $25,000,000 per repository. The total decommissioning costs, excluding room backfilling, are shown in Table 5.6.2 for spent fuel and reprocessing-waste repositories.

TABLE 5.6.2. Decommissioning Costs for Spent Fuel and Reprocessing-Waste Repositories

Repository Media	millions of dollars	
	Spent Fuel Repository	Reprocessing-Waste Repository
Salt	50	55
Granite	50	54
Shale	50	54
Basalt	50	55

5.6.4 Unit Costs

Levelized unit costs are calculated charges per unit of production sufficient to recover all construction costs, including interest, and to pay all operating and decommissioning costs. For this study, the weighted cost of capital for the Federal government is assumed to be 7% but a range of 0 to 10% was utilized to develop uncertainty estimates. Additional information on the calculation of unit costs can be found in DOE/ET-0028, Vol. 1, Section 3.8.5.

The levelized unit costs for waste isolation in geologic repositories, based on the conceptual repositories used in this Statement, are shown in the last column of Table 5.6.1. These costs are expressed in dollars per kilogram of heavy metal of isolated spent fuel for spent fuel repositories or in dollars per kilogram of heavy metal reprocessed for reprocessing-waste repositories. Isolation in salt repositories costs significantly less than isolation in any other medium for either waste type with the exception of isolating spent fuel in shale. Shale is the next least expensive medium for disposing of either spent fuel or reprocessing cycle wastes. Granite is the next least expensive and basalt is the most expensive medium for isolating wastes. Unit cost differences between repositories storing spent fuel and repositories storing reprocessing waste (in the same geologic medium) do not appear to be significant, with the possible exception of repositories in shale. Because of the preliminary nature of the conceptual designs, uncertainty in the mining procedures and in the cost of money, the overall uncertainty in the total unit cost estimates is estimated to be +50%.

A breakdown of the unit costs for waste disposal by waste type for the reprocessing cycle wastes is shown in Table 5.6.3 for each of the four geologic media considered here.

TABLE 5.6.3. Unit Costs by Waste Type and Geologic Media

Waste Type	$/kgHM			
	Salt	Granite	Shale	Basalt
HLW	24	39	41	51
RH-TRU Canisters	3	5	4	5
RH-TRU Drums	18	29	24	32
CH TRU	2	4	4	5
Total	47	77	73	93

5.6.5 Comparison with Other Cost Data

Recent repository cost estimates, including the estimates in this statement, use as their principal basis one of three independent repository conceptual design studies (Kaiser 1978, Stearns-Rogers 1979, OWI 1978). The Bechtel (1979) spent fuel disposal study uses the conceptual designs reported for Kaiser (1978) and Stearns-Rogers (1979) with variations based on differences in waste form. The repository cost included in DOE's preliminary spent fuel acceptance charge estimate DOE/ET-0055 (DOE 1978) is based on a planning study by Kaiser Engineers prior to the completion of their conceptual design estimates. The recent Environmental Impact Statement on Spent Fuel Policy, DOE/EIS-0015 (DOE 1980a), uses this same basis. The estimates in this Statement are based on OWI (1978) with design modifications as noted in Section 5.3.

The capital cost estimate for spent fuel repositories given in Bechtel (1979), DOE (1978), and DOE (1980a) is $500 million with annual operating costs of about $50 million. The main difference between these estimates and those in Table 5.6.1 is that a portion of the mining cost is allocated to operating cost instead of being totally included in the construction cost. The unit cost calculation for spent fuel disposal in a bedded salt repository of $51/kg heavy metal in DOE/ET-0055 compares favorably with the $52/kg calculated in Table 5.6.1 (both costs are in 1978 dollars).

In the DOE Statement of Position to the NRC Rulemaking Proceedings (1980b), cost estimates are given for spent fuel disposal in salt, granite and basalt media. Total capital, operating and decommissioning costs of $2.2 billion ($1.8 billion in 1978 dollars) for a bedded salt repository are in general agreement with this Statement. However, total costs for granite and basalt repositories reported in DOE (1980b) are about $2 billion less than estimated here since the standardized mine layouts used in the DOE (1980b) estimate postulate substantially less rock removal per unit of waste emplaced than does this Statement.

5.6.6 Other Cost Considerations

Costs associated with the retrieval of spent fuel elements from the repository during the 5-year retrievable period, subsequent interim storage at the repository site and transportation to a new site are estimated to be no more than the figures presented in Table 5.6.4.

TABLE 5.6.4. Spent Fuel Retrieval Costs

	$/kgHM			
	Salt	Granite	Shale	Basalt
Retrieval	14	18	15	18
Interim Storage	22	22	22	22
Shipment to New Repository (~1500 mi)	32	32	32	32
Total	68	72	69	72

The disposal costs given in Table 5.6.1 apply for all cases in which spent fuel disposal requirements are at least equivalent to 48,000 MTHM. For the case in which disposal requirements are limited to the 1980 inventory of spent fuel (about 10,000 MTHM), unit repository costs would be approximately:

	$/kgHM
Salt	90
Granite	100
Shale	90
Basalt	110

The total costs of waste management including disposal are presented and compared to the total cost of electric power production in Section 7.6.

REFERENCES FOR SECTION 5.6

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U.S. Department of Energy. 1980b. Proposed Rulemaking on the Storage and Disposal of Nuclear Waste (Confidence Rulemaking)--Statement of Position of the United States Department of Energy, DOE/NE-0007, Washington, D.C.

5.7 SAFEGUARDS INCLUDING PHYSICAL PROTECTION FOR GEOLOGICAL DISPOSAL

Facilities associated with geologic repositories will employ safeguards and physical protection measures commensurate with the potential risks to society associated with the waste material (see discussion in Section 4.10), and the surface facilities at these sites would receive the principal emphasis. After emplacement in the geologic repository, the spent fuel and wastes would be very inaccessible for theft or diversion. Sabotage, if achieved, would have a minimum effect on the safety and health of the public because of the containment of the waste in a solid material that is difficult to pulverize and disperse. Nevertheless, sabotage of the facility and the waste packages must be guarded against until repository closure.

5.7.1 Geologic Disposal of Spent Fuel

Safeguards, including physical protection measures afforded vital material, would be required for the spent fuel elements as they are received, inspected, and made ready for geologic placement. This material is not attractive for theft or sabotage for the reasons given previously (Section 4.10.1.2), and in addition it becomes more inaccessible at this facility. Moreover, the currently required physical protection measures include controlled access through two barriers plus an adequate security force, and a contingency plan (response force) as required by the Federal regulations (10 CFR 73). Records of waste disposition to provide traceability from origin to final disposal will be maintained (43 CFR 195 1978).

After emplacement and closure in the geologic repository, the spent fuel would be essentially inaccessible for sabotage or theft. A successful intrusion and theft of HLW containers or sabotage in place would be unlikely because of the limited access to the containers, the operational control over entry, and the physical security provided at the access points in the surface facility. After repository closure the waste would be available only through re-excavation or mining. Theft or sabotage after closure and decommissioning does not appear credible because the effort would be readily detectable.

5.7.2 Geologic Disposal of Solidified High-Level Waste and Transuranic Wastes

The physical protection required for the surface facility handling these wastes includes measures to protect the facility and material from intrusion, theft and sabotage. These measures would be similar to those in any facility handling moderately hazardous material. At the repository these materials would be quite inaccessible to the public, and in a form that is not attractive for theft or sabotage. The solidified high-level waste would be too radioactive for adversaries to handle except remotely behind heavy shielding which, as shown in earlier discussions, makes this material inherently unattractive. Routine accountability programs would record the transfer of this material to its geologic disposal location. After geologic emplacement this material would be relatively inaccessible for theft. Sabotage, if ever attempted, would have little affect on the public because of the containment of the waste. After closure, theft or sabotage does not appear credible because mining or re-excavation would be required to gain access. Such an operation would be difficult to conceal and could be easily prevented.

REFERENCES FOR SECTION 5.7

Code of Federal Regulations. Title 10, part 73.

"Storage of Spent Fuel in an Independent Spent Fuel Storage Installation." 1978. Federal Register, Vol. 43, p 195.

5.8 IRREVERSIBLE AND IRRETRIEVABLE COMMITMENT OF RESOURCES ASSOCIATED WITH GEOLOGIC REPOSITORIES

Resources that will be irretrievably committed in disposal of radioactive wastes in geologic repositories are the energy resources consumed in repository construction and operation, cement (a relatively energy intensive material in concrete) and any canister or engineered barrier materials committed to the repository with the waste. Ranges of commitments of these resources for the several geologic disposal media, on a normalized energy production basis of one GWe-yr, are presented below:

	Spent Fuel Repositories	Fuel Reprocessing Waste Repositories	Approximate U.S. Annual Production
Propane, m^3	1.6 - 1.9	1.4 - 3.1	1×10^6
Diesel Fuel, m^3	$1.1 \times 10^2 - 1.8 \times 10^2$	$1.6 \times 10^2 - 2.2 \times 10^2$	4×10^8
Gasoline, m^3	$1.2 \times 10^1 - 1.4 \times 10^1$	$1.1 \times 10^1 - 2.2 \times 10^1$	6×10^8
Electricity, kw-hrs	$9.9 \times 10^5 - 1.3 \times 10^6$	$1.2 \times 10^6 - 1.8 \times 10^6$	2×10^{12}
Manpower, man-yrs	$2.1 \times 10^4 - 2.9 \times 10^4$	$3.2 \times 10^4 - 5.3 \times 10^4$	4×10^6 (a)
Steel, MT	$1.8 \times 10^1 - 2.8 \times 10^1$	$6.1 \times 10^1 - 8.1 \times 10^1$	1×10^8
Cement, MT	$2.1 \times 10^1 - 2.7 \times 10^1$	$3.1 \times 10^1 - 4.4 \times 10^1$	7×10^7
Lumber, m^3	1.8 - 2.1	2.4 - 3.3	3×10^9

(a) Construction and mining.

Even at an installed nuclear power capacity of 250 GWe operating over several decades the above material and energy commitments are but a small fraction of that used for the total economy. To give additional perspective to the consumption of energy, fossil fuels, and electrical consumption, each were converted to units of energy expended in deep geologic disposal of waste per unit of energy produced by the fuel from which the waste came. In the case of spent fuel 0.04% of the energy produced was consumed in geologic waste disposal and in the case of fuel reprocessing wastes 0.05% of the energy produced was consumed. On the above bases it is concluded that the irretrievable commitment of the above materials is warranted.

5.9 SHORT-TERM USES OF THE ENVIRONMENT VERSUS LONG-TERM PRODUCTIVITY

In terms of short-term use, about 800 ha (2000 acres) will be restricted from present use and until the repository is decommissioned (on the order of 30 years). After decommissioning, this land could be returned to its former use. An exception would be the area on which excess rock had been stockpiled, assuming no use elsewhere had been found for the rock. The area that this rock would cover would depend on the height to which it was finally piled. Characteristics of specific sites would probably dictate the size and shape of the rock storage pile(s). If the height of the storage pile were about 3 m (10 ft) the pile (ignoring the angle of repose of the rock) would occupy an area about 2200 m (1.4 miles) on a side. If left in this state, this large pile would constitute a cost in terms of lost productivity of the surface soils and in terms of an aesthetically displeasing visual impact. On the other hand, this large pile for granite and basalt repositories could be moved and modified to form a suitable marker for the repository. The costs would be balanced by the benefits of permanent isolation of radioactive waste far beneath the earth's surface.

5.10 UNAVOIDABLE ADVERSE ENVIRONMENTAL IMPACTS ASSOCIATED WITH RADIOACTIVE WASTE DISPOSAL IN GEOLOGIC REPOSITORIES

Impacts associated with nonradiological accidents during construction of geologic repositories and the dose to the work force emplacing the wastes, are perhaps the most significant unavoidable adverse impacts. In the strictest sense, such accidents should be avoidable, but experience in construction and mining suggests they will happen even with the best safety programs. The estimated number of expected fatalities (or permanent disabling injuries) ranged from 6 to 17 per 1000 GWe-yr of electrical energy generation, depending on repository media and whether disposal is for spent fuel or for fuel reprocessing wastes. While the number of lives which might be lost during mining operations could be obviated by some other disposal alternative, the radiation dose from waste disposal would be comparable (at least at this stage of estimating) for alternative disposal methods. (As a point of perspective, about 200 linemen would be expected to lose their lives in the process of bringing 1000 GWe-yr of electrical energy to its users regardless of the generation mechanism.)

The radiation dose to the work force emplacing the waste was estimated to be 4×10^3 man-rem for spent fuel and 8×10^4 man-rem for fuel reprocessing wastes for 1000 GWe-yr of electrical energy production. Using the conversion of 50 to 500 fatal cancers per million man-rem, about 2 radiation-related fatalities would be expected for emplacement of spent fuel; and 4 to 40 from emplacement of fuel reprocessing wastes for 1000 GWe-yr.

Radiation dose to population groups was not significant even in the case of postulated accidents during repository operation. Hazards to workers from potential operational accidents (canister drop down the mine shaft) were found to be very serious; however, additional safety features as suggested would reduce the risk substantially. For disruptive events in the long term the societal risk from wastes disposed of in geologic repositories was found to be small in comparison to societal risks such as from lightning strikes.

Adverse impacts on the terrestrial and aquatic environments could result from inadequate precautions taken for management of mined rock stockpiled on the surface, particularly in the case of repositories in salt and to a lesser extent in the case of repositories in shale.

The potential for boom/bust socioeconomic problems was determined to be very high for sites that may be isolated from needed labor pools. Although highly site specific, plans to lessen or obviate socioeconomic impacts are likely to be required for the site selection process.

There will likely be adverse psychological impacts among some members of the public because of the presence of a repository in their locality. A program to explain the exact nature of the repository facility and the multiple features present to prevent release of radioactive materials could lessen the concerns of the local public as long as information is completely presented and the activities of DOE are open to the scrutiny of local community leaders.

Chapter 6

ALTERNATIVE CONCEPTS FOR WASTE DISPOSAL

A number of possible alternative methods for the disposal of nuclear waste have been suggested. These concepts have been evaluated and developed to various degrees by different organizations. The status of technology is described in this section, as are advantages and disadvantages of each concept. The intent is to address the various concepts as consistently as possible to facilitate the comparison of the potential impacts of their implementation.

The alternative concepts discussed are: the very deep hole, rock melting, island repository, subseabed, ice sheet, well injection, transmutation, and space. These are all compared to the mined repository concept.

6.1 PRESENTATION/ANALYSIS OF ALTERNATIVE DISPOSAL CONCEPTS

This section presents concept descriptions and discussions of potential health and environmental impacts for eight radioactive waste disposal methods that have been suggested as alternatives to disposal in mined geologic repositories. These presentations are based on sections from the draft of this Statement, updated to incorporate current information resulting from continuing development and evaluation of alternative concepts. Information presented here is taken from available results of relevant studies. References, cited throughout the text to indicate sources of significant parameter values and statements, are listed at the end of subsection 6.1. In addition, bibliographies are provided in Appendix M to indicate other information sources for each concept. The concept descriptions are also supported by information in Chapters 3, 4, and 5 of this EIS and reference is made to those chapters as appropriate.

The discussion of each disposal concept covers the following topics:

- Concept Summary
- System and Facility Description
- Status of Technical Development and R&D Needs
- Impacts, Both Preemplacement and Postemplacement
- Cost Analysis
- Safeguard Requirements.

Because concept descriptions, environmental impacts, and estimated costs for each option were taken from various sources that used different basic assumptions, the information provided here for each concept is not normalized to a standard set of conditions, e.g., a common annual throughput or a common environment. As an example, the well injection concept section presents radiological impact information extracted from a reference which addresses the impacts of intermediate level waste disposal. This is done to provide the reader with related information that may be important to the understanding of the concept. In addition, the space disposal and transmutation concepts require chemical processing of spent fuel to prepare waste for disposal or elimination. Accordingly, comparisons between these concepts and, for example, others not requiring processing would be difficult. For instance, transportation costs in the processing case could not be compared with those for disposal of spent fuel.

Four of the concepts (very deep hole, rock melt, space, and subseabed), however, were covered in a common reference and thus have a common basis. The other options are not normalized because, for example, while linear extrapolation to a higher or lower quantity of waste handled may result in a more or less conservative estimate of impacts and costs for a particular option, it may also bias any comparative analysis for or against that concept. Also, the descriptions, impacts, and costs that have been reported for some of the alternatives are incomplete because of the early stage of the alternatives' technical development.

In addition to being, in many cases, incomplete, the cost and impact data should be considered speculative. For example, the costs projected for the development of an alternative are generally based on judgment regarding the current state of technical uncertainty for the alternative. In practice, many such cost estimates do not adequately anticipate the expanded scope of activities that may result as additional uncertainties and issues are identified in attempts to resolve the original set of uncertainties. It was felt, therefore, that manipulating costs and impact information may indicate more significance than is warranted.

The disposal methods along with rates used as a basis for defining each of the alternatives, including the mined geologic repository, are:

Alternative	Disposal Rate, MTHM/yr	Reference
Mined Geologic Repository	6,000	Chapter 3
Very Deep Hole	5,000	Bechtel (1979a)
Rock Melt	5,000	Bechtel (1979a)
Island	Disposal rates similar to mined geologic repository. Ocean transportation similar to subseabed concept, see section 6.1.	Chapter 5, and Section 6.1.4
Subseabed	5,000	Bechtel (1979a)
Ice Sheet	3,000	MITRE (1979)
Well Injection	Unspecified	ORNL TM 1533, DOE (1979)
Transmutation	2,000	Blomeke et al. (1980)
Space	5,000	Bechtel (1979a)

Frequently, numbers taken from the various references are rounded to an appropriate number of significant digits in an effort to simplify this section of the document.

The general approach to each of the topical discussions used to describe the alternatives is as follows.

Concept Summary. The concept summary provided for each alternative contains a general discussion of the disposal concept, highlights significant technical aspects of the concept, and establishes a basis for specific system and facility descriptions, technology status, and environmental impact analyses.

System and Facility Description. In this section, the systems and facilities associated with a reference repository system design for each alternative disposal concept are described. Each description begins with a discussion of the fuel-cycle options reflected in the reference system design. The options and the selections made are illustrated by a standard diagram.

The waste-type compatibility for each concept is discussed, providing a basis for defining waste types that can and cannot be accepted by the disposal system. This section also indicates if the total fuel cycle involves chemical processing and if there is a need for a mined geologic repository (or other additional facility) to accept some portion of the waste.

The waste management system descriptions cover predisposal treatment and packaging (with reference to Chapter 4), surface facilities and equipment, and transportation systems. These descriptions vary substantially because of differences among the alternatives, e.g., space disposal compared to transmutation. System descriptions provide a basis for subsequent discussion of technology status and R&D requirements, potential environmental impacts, and cost analysis.

Status of Technical Development and R&D Needs. This section provides an insight into the technical status and R&D needs associated with the development of each disposal option. The discussions are based on the most current reports contained in the large body of references available for disposal options. Emphasis was placed on documents prepared by organizations that have played a definitive role in the development or evaluation of specific options.

Each disposal option is at a different stage of development ranging from ice sheet and rock melt, which are in only the early conceptual stage, to well injection, which has been used for the disposal of remotely handled waste at the Oak Ridge National Laboratory. Wide disparity in the states of development, however, should not be used to connote the degree of difficulty anticipated in deploying a particular option.

Current technological issues unique to each option are identified. These issues depend on the state of development. As knowledge is accumulated and refined on a specific concept to resolve technical issues, it may often reveal additional technological concerns to be resolved.

Specific research and development requirements ascribed to each disposal option are those contained in references provided by organizations involved in the development or evaluation of the particular disposal option. The requirements identified are based on technological issues and programmatic needs.

Estimates for implementation time and research and development costs depend on the degree of planning information available for the disposal concept. For example: no estimates are identified for well injection because of lack of definitive program plans. Available estimates for space disposal go through concept definition and evaluation only. Estimates for ice sheet disposal, however, include all of the currently anticipated activities required to develop and implement an operational system.

Impacts. Impacts are presented on the basis of information found in the reference material. Impacts for these sections are separated into Health Effects Impacts (the human environment) and Natural System Impacts. Natural System Impacts include impacts to ecological and geologic/hydrologic systems. The term Natural System Impacts therefore includes impacts other than those to the human environment. The reader is cautioned that for those alternatives that are more advanced in their technical development, a greater number of environmental impacts are identified. Likewise, for those disposal methods that are in a preliminary stage of development, there may be other environmental impacts that have not yet been determined.

In general, the methodology followed in calculating impacts is not described, but reference is made to original material where the reader can find this information. No attempt has been made to develop a common impact assessment methodology, so the methods applied vary from study to study. For these reasons, the values presented are not always comparable on a one-to-one basis. It is believed, however, that sufficient information is provided to allow a qualitative comparison of the alternatives (see Section 6.2).

Cost Analysis. The cost analyses provide capital, operational, and decommissioning cost estimates based on information available from references authored by organizations involved in the evaluation or development of the specific disposal options. The costs are those directly attributable to the disposal mode under consideration and not on support modes such as waste preparation or routine transportation. All cost estimates are given in 1978 dollars, derived by an adjustment of 10 percent per year of estimates based on non-basis years.

The reader is cautioned about the preliminary nature of the cost estimates. Also, in many cases, due to the underdeveloped status of most of the alternatives, full cost data are not available. In such cases only referencable information is presented. No attempt is made to estimate system or component development, capital, operating or decommissioning cost where these, estimates could not be based on open literature reference. For example, in the case of the transmutation concept, a comprehensive and conclusive fuel cycle cost analysis has not been performed such that an aggregate cost estimate could be prepared. In addition, the impacts to the costs of disposal of the residual wastes from the transmutation concept are not known.

The estimates do not include transportation and waste-form preparation costs associated with the disposal method. However, unique transportation and waste-form requirements, in addition to the need for supplemental storage or disposal, are identified.

The cost analyses for very deep hole, subseabed, rock melt, and space disposal are based on estimates contained in a current reference that used consistent waste disposal rates over the same time period. The available costs for the other disposal options, including the mined geologic repository, are not normalized to the same waste disposal scenario. Cost estimates are sufficiently accurate, however, for a qualitative comparison.

Safeguard Requirements. In this section, the vulnerability of each alternative waste disposal concept to the diversion of sensitive materials or terrorist acts of sabotage is qualitatively discussed. In addition, the features unique to the alternative that enhance or detract from that vulnerability are described. For more detailed discussion of safeguards for predisposal operations the reader is referred to Section 4.10.

6.1.1 Very Deep Hole

6.1.1.1 Concept Summary

The very deep hole (VDH) concept involves the placement of nuclear waste as much as 10,000 m (32,800 ft) underground, in rock formations of high strength and low permeability. In this environment, the wastes might be effectively contained by the distance from the biosphere and the location below circulating groundwater as they decay to innocuous levels (OWI 1978 and ERDA 1978). To act as a nuclear waste repository, the host rock would have to remain sealed and structurally stable under the heat and radiation introduced by the wastes. Potential rock types for a repository of this kind include crystalline and sedimentary rocks located in areas of tectonic and seismic stability.

An immediate question concerning this concept is: "How deep is deep enough?" The required depths would place the wastes far enough below circulating ground waters that, even if a connection develops, transport of materials from the repository to the surface would take long enough to ensure that little or no radioactive material reaches the biosphere (LBL 1979). The absolute value of this depth is not yet determined.

Defining the necessary depth at a given site requires determining site-specific limits on the transport of radioactive materials to the biosphere, the site-specific hydrologic regime, and the heat-source configuration (waste packing). Available data from the literature, primarily from the oil and gas industry, show that some sedimentary rocks are porous and permeable and may contain circulating groundwater to depths in excess of 9,000 m (30,000 ft). Investigations of crystalline rock, although very limited, suggest that at much shallower depths some such rocks have relatively low porosities and permeabilities. Hence "very deep" for these crystalline rocks may mean just a few thousand meters instead of the 9,000 m or more required for sedimentary rocks. Once the required depth has been determined, the technology for making the hole to that depth and the ability of the surroundings to accept the heat source become the limiting factors. It is clear that problems of making the hole, holding it open, emplacing the waste, and sealing the hole must be considered together. Should shallow depths be determined as adequate, many of the potential problems of the very deep hole concept (e.g., drilling technology and ambient conditions at depth) would be mitigated.

The concept assumes that disposal in very deep holes would not permit retrieval of wastes. It would also provide assurance that no climatic or surface change will affect disposal.

Environmental impact considerations for the very deep hole concept are those associated with drilling a deep well or sinking a deep shaft, constructing the predisposal surface facilities, emplacing the wastes, decommissioning the facilities, and ensuring long-term containment of the wastes.

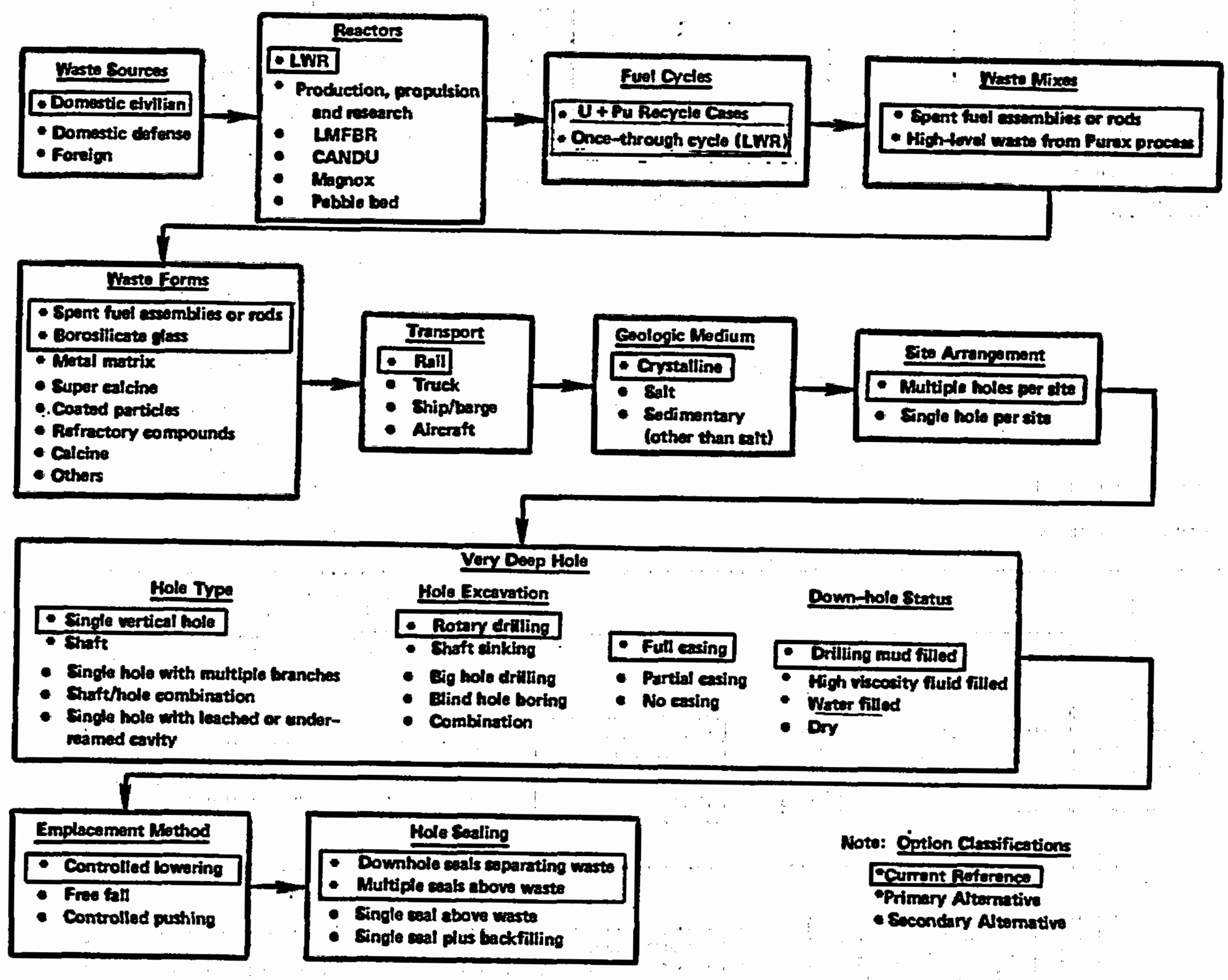


FIGURE 6.1.1. Major Options for Very Deep Hole Disposal of Nuclear Waste

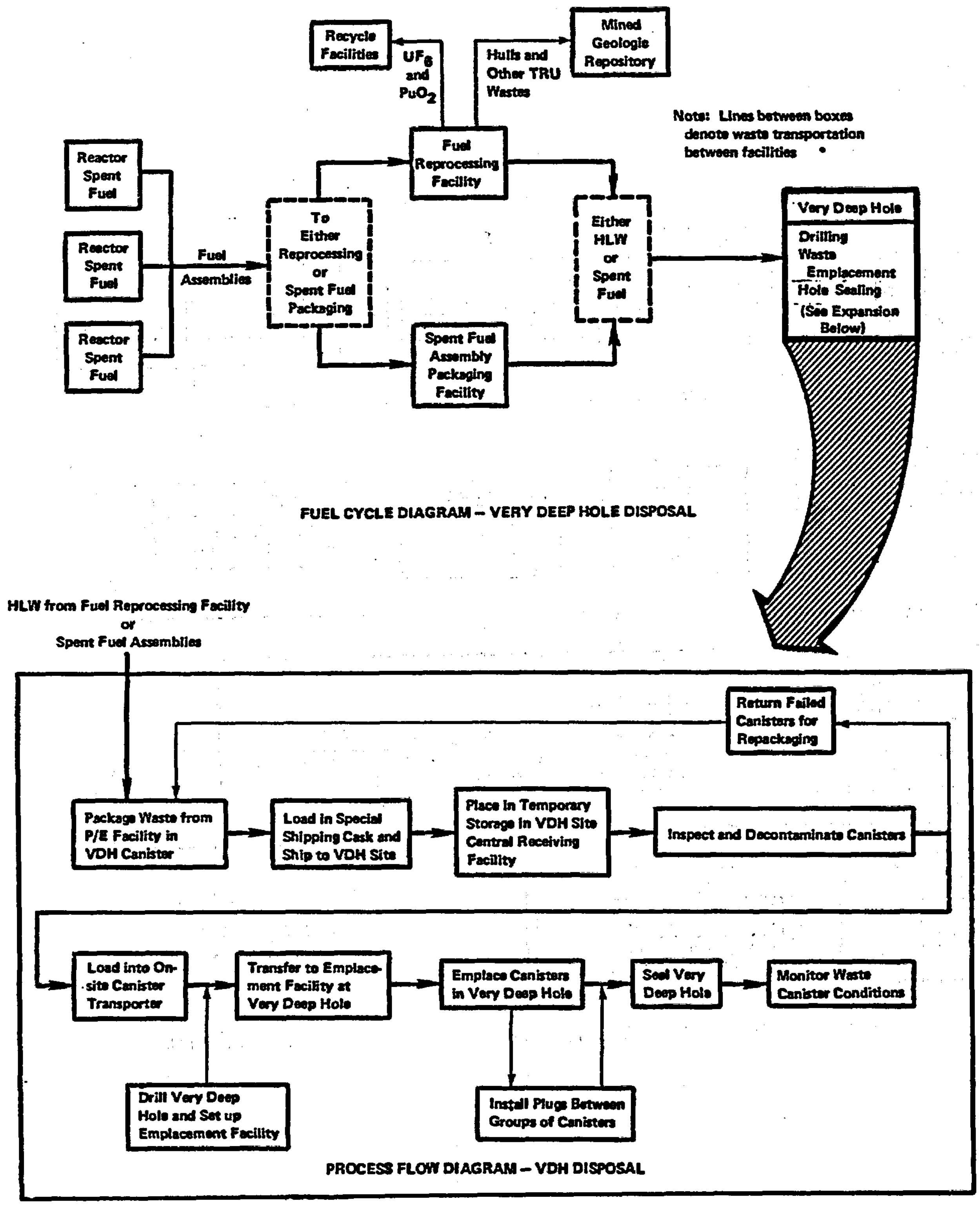


FIGURE 6.1.2. Waste Management System--VDH Disposal

6.1.1.2 System and Facility Description

System Options

The reference concept for the initial VDH disposal of nuclear waste has been developed from a number of options available at each step from the reactor to disposal in the very deep hole.

Various options to be considered for VDH disposal are summarized in Figure 6.1.1. The bases for selection of options for the reference concept (those blocked off) are reviewed in detail in various documents listed in Appendix M.

Because options for the waste disposal steps from the reactor up to, but not including, the geologic medium are similar for mined geologic repositories and VDH disposal, the options selected for the reference design are similar for the two concepts. From that point on, the options selected for the reference design are based on current program documentation for VDH disopsal.

Waste-Type Compatibility

Very deep hole disposal would be limited to unreprocessed spent fuel rods and the HLW from uranium-plutonium recycle cases. Because of cost constraints, VDH disposal of contact handled and remotely handled TRU wastes is not considered likely. Handling the large volume of these wastes would substantially increase drilling activities, costs, and the extent of adverse environmental impacts for VDH disposal. Thus, the low- and intermediate-level TRU wastes would require some other form of terrestrial disposal. It is assumed for the reference case that these wastes would be placed in mined geologic repositories.

Waste-System Description

The reference concept design was selected through judgment of a "most likely" approach based on available information and data. The fuel cycle and process flow for the reference concept are shown in Figure 6.1.2. In the reference concept, a VDH repository is designed for disposal of 10,200 canisters per year of spent fuel or for 2,380 canisters per year of solidified HLW. With a 40-year repository operation period, emplacement of spent fuel would require 68 holes per year with 150 canisters placed in each. Multiple holes would be drilled while others are being filled. HLW would require emplacement of 375 canisters per hole in six to seven holes per year (Bechtel 1979a), also with simultaneous drilling and emplacement operations.

Predisposal Treatment and Packaging. The predisposal treatment of waste for the VDH concept would be identical in many respects to the predisposal treatment of waste for the mined geologic repository concept. Chapter 4 of this document discusses the predisposal systems for both spent fuel and HLW common to all of the disposal concept alternatives.

The specific waste form required for emplacement in the deep hole is not yet identified. The waste form and canister would have to be structurally strong to resist downhole stresses and crushing forces, and chemically resistant to the waste emplacement medium. A metallic matrix or a granular waste form would be possible (Bechtel 1979a).

The canister would have to provide for safe handling, shipping, and emplacement of the waste. Both the HLW and the spent fuel canisters would have to be packed solidly to avoid crushing due to hydrostatic pressure of drilling "mud" (lubricant) left in the hole to counter lithostatic pressure. The canisters and spacers would have to be dense enough to sink through the mud slurry to the bottom of the hole. Carbon steel is considered as one candidate canister material that will fulfill these requirements (Bechtel 1979a). However, more complex designs using multiple barriers may be required.

The canisters for both HLW and spent fuel would have to be small enough for emplacement in a hole lined with a steel casing. HLW canister dimensions identified for the reference case accommodate the fuel. Dimensions identified for the reference case are 36 cm (14 in.) diameter and 4.6 m (16 ft) long (Bechtel 1979a and TID 1978).

Site. The critical geologic parameters that will determine the feasibility and impact of nuclear waste disposal in a deep hole system and that must be considered in site selection are:

- Lithology
- Tectonics and structural setting
- Hydrologic conditions
- States of stress
- Mechanical properties of the rocks at depth
- Natural thermal regime
- Geochemical reactions.

The interactions of these parameters and the effect of heating by the waste (thermomechanical factors) may also be significant. Geologic assumptions underlying the VDH concept are that the hole will be drilled, or a shaft excavated, in a regime of moderate to low geothermal gradient in rock with high strength and low permeability. Furthermore, the wastes are to be deposited irretrievably - not stored (LBL 1979). The specific geotechnical considerations are addressed in detail in LBL (1979) and Brace (1979).

Since more holes would be needed, emplacement of spent fuel during a 40-year period would require a total land area of approximately 140 km^2. Canisters would be shipped by rail from a processing and encapsulation facility to the repository site, which would consist of a number of drilled holes around a centrally located receiving facility (Bechtel 1979a).

Waste Receiving Facility. The central waste receiving facility at the deep hole site would be used to unload the waste canisters, store them temporarily, and perform any work required to assure prompt emplacement in the hole. The receiving building would contain a cask handling area, a canister storage area, a hot cell, and auxiliary facilities (see Bechtel 1979a).

The cask handling area would contain facilities for receiving, cleaning, and decontaminating shipping casks and for reloading empty casks on rail cars. Upon arrival, an overhead bridge crane would remove the loaded shipping cask and move it to the confinement section of the building. The lid would be removed and the cask aligned with a hot cell port. The HLW or spent fuel canisters would be removed remotely to a storage rack within the hot cell.

An interim dry storage area adjacent to the hot cell would have space for a 1-month supply of canisters.

The hot cell would include space for checking the canisters for visible damage, radiation leakage, and surface temperature. Facilities would be provided to decontaminate waste handling equipment in case of a canister failure. Damaged canisters would be overpacked and returned to the processing and emplacement facility for repacking.

The receiving facility would also provide auxiliary services such as ventilation, equipment maintenance, and a control system.

Canister Transporters. Canister transporters, similar to those used for subsurface transportation and emplacement in the mined geologic repository (Section 5.4), would be used to transfer the waste from the receiving facility to the emplacement facilities. Each transporter would consist of a wheeled vehicle suitable for operation on site roadways, a shielded transfer cask, and equipment for raising and lowering canisters in and out of the transfer cask. In the receiving facility, the transporters would be positioned over a portion of the hot cell to bottom load the canisters into the transfer cask. At the emplacement facility, the transporters would be positioned over the temporary storage area and the canisters would be bottom discharged into temporary storage.

Drilling System. The drilling rigs would be similar to those used in the gas and petroleum industries and would be portable for movement from one hole location to another on the site. Each complete rig would require a clear, relatively flat area, approximately 120 x 120 m (400 x 400 ft), at each hole location (McClean 1977).

In the reference concept, the drilled hole for spent fuel is 48 cm (19 in.) in diameter and 10,000 m deep (Bechtel 1979a). For HLW, the hole is 40 cm (16 in.) in diameter. The depth and diameter, however, could vary depending on the geologic medium, the depth required to satisfy containment requirements, and the drill rig capacity. For HLW, the hole would be fully cased to the required depth with seamless steel pipe about 40 cm in outside diameter, which would reduce the hole diameter available for waste.

Oil field rotary drilling techniques would be used to sink the holes, which may be stepped down in diameter as the depth increases. To seal the pipe to the rock, a grout would be forced through the pipe and then back up between the wall of the hole and the outside of the casing. The bottom of the hole would be sealed.

During the drilling and emplacement operation, the hole would be kept full of drilling mud to facilitate drilling, prevent casing and canister corrosion, minimize casings sticking to the sides of the hole during installation, and counter lithostatic pressure.

Emplacement Facilities. Each emplacement facility would include a confinement enclosure to provide a controlled environment for emplacement operations, and the temporary canister storage facility (Bechtel 1979a). The entire emplacement facility would be on rails for movement from hole to hole on the site.

As described above, canisters would be transferred from the receiving facility to the temporary storage facility, which would provide shielding and an accumulation area for canisters to accommodate differences between transfer and emplacement operations. Emplacement equipment with cable totaling at least 10,000 m in length would lift a waste canister from temporary storage into a shielded cask, position it over the very deep hole, and lower it through the bottom of the cask into the hole (Bechtel 1979a). The waste canisters would be lowered into the lower 1,500 m (5,000 ft) of the hole with metallic honeycomb spacers placed between each canister to absorb impact in case a canister is dropped (Bechtel 1979a). If required by canister structural design limits, a structural plug, anchored to the sides of the hole, would be emplaced between groups of canisters to support the load.

Sealing Systems. After all waste canisters are in place, the hole would be sealed to isolate the waste from the biosphere. Sealing could include plugging both the hole and the damaged rock zones around the hole.

The components of the sealing system would have to have low permeability to limit nuclide migration and sufficient strength to maintain mechanical integrity for a specified period. Possible plugging materials include inorganic cements, clays, and rock. The specific material or materials would be selected for compatibility with the geologic medium and down-hole conditions (Bechtel 1979a). Plugging could be done with standard equipment typically used by a drilling rig crew. For final sealing and closure of the very deep hole, drill rigs, similar to those described for hole drilling, would be set up at the hole location.

Retrievability/Recoverability. Waste canisters would be retrievable as long as they are attached to a cable during the emplacement process. Once the canister is disengaged, it would become essentialy irretrievable. Post-enclosure recovery is likewise considered nearly impossible.

6.1.1.3 Status of Technical Development and R&D Needs

Present State of Development

The status of equipment facility, and process development for different operational phases of VDH emplacement are considered below.

Drilling Techniques. Four methods to excavate a very deep hole have been considered. These are oil field rotary drilling, big hole drilling techniques, drill and blast shaft sinking, and blind hole shaft boring. The latter three methods are limited in the depths that can be attained at present and in the foreseeable future. They might have applications in specific geologic media but will not be considered further here since the possibility of their use appears remote for waste emplacement in this concept. For details on these concepts, see LBL (1979).

For oil field rotary drilling, standard oil field drill equipment would be used. In this method, a drill bit attached to a drill pipe is rotated from the surface, and drilling mud is circulated through the drill pipe to carry cuttings to the surface. The drilling mud also assists in providing borehole stability, provides lubrication and cooling, and minimizes pipe sticking. Substantial rotary drilling experience exists; however, most of the drilling has been in sedimentary formations.

At least the upper portions of deep rotary drilled holes would be cased; and, in fact, the entire hole may need to be cased for borehole stability, as in the reference concept (LBL 1979). As described there, cement grouts would be pumped from the bottom of the hole up around the steel casing to seal the casing against the drilled borehole. If the entire borehole were cased, then the hole could be bailed dry (depending on the depth of the hole), and could be left standing open for extended periods. If the bottom portion of the hole were not cased, it is unlikely that the borehole would stay open if the hole were bailed dry. Some fluid, probably with a density somewhat higher than that of fresh water, would therefore be required in the open hole at all times.

There is little experience at drilling in hard, crystalline rocks, although such rocks may pose no more, or fewer, problems than drilling ultra-deep wells in sedimentary rocks. A limited number of oil field rigs are capable of drilling to 8,000-m (25,000 ft) depths and beyond, and there are presently four rigs in the U.S. capable of drilling to a depth of 9,000 m. The bottom portions of such holes have been drilled with a 16.5 cm (6-1/2 in.) diameter bit, and the holes were cased to the bottom. There is some experience in drilling geothermal wells where formation temperatures are 30-0 C (approximately 600 F) as anticipated in VDH drilling; however, these wells have not been drilled much below 3,000 m (10,000 ft).

It is believed that deeper and larger diameter holes could be drilled. A maximum well depth of about 11,000 m (36,000 ft) in rocks where borehole stability is not a problem is believed possible, using a 20-cm (7-7/8 in.)-diameter bit for the bottom hole. Depths of 9,000 m could be achieved with 31-cm (12-1/4 in.)-diameter bits in crystalline rocks where no gas pressure exists. For very strong rocks, the bottom part of the hole might be left open. In fact, for the 31-cm-diameter hole, the bottom part of the hole may have to remain open because current rigs (with current casing) would not be able to set casing to the bottom of a 9,000 m hole. A drill rig with a 15,000-m (50,000-ft)-depth capability has been designed but not operated which would utilize the largest available components. It would provide a 22-cm (8-1/2-in.)-diameter hole at total depth (Drilling DCW 1979). Salt has been drilled successfully to about 4,600 m (15,000 ft); below this, borehole closure prohibits further drilling.

Emplacement. The technology for emplacing waste canisters is not fully developed at present. Some technology for emplacing items to depths less than 10,000 m exists. For example, the Deep Sea Drilling Project has a hydraulically operated down-hole device that disconnects the boring bits.

Sealing. Standard oil field practices for cementing in casing have satisfactorily isolated deep high-pressure gas zones from shallower formations and from the surface for time periods measured in decades. Plugs of cement or other materials have been emplaced in abandoned oil and gas wells, both cased and uncased, and have maintained integrity over similar periods of time. In these instances, it is not uncommon for the casing to corrode prior to plug deterioration.

Logging/Instrumentation. Borehole geophysical logging techniques in existence and currently under development will permit the logging and analyses of a number of parameters critical to the emplacement of radioactive waste in very deep holes. Caliper, acoustic, televiewer, and other borehole geophysical devices are regularly used to verify the presence and distribution of fractures in well bores. Electrical logs, neutron porosity loss recorders, and other devices are used to verify the presence of water. Temperature logs and spinner logs are used to detect water flow. While all of this equipment can be used from depths of hundreds to thousands of feet, none of these tools can function at the temperatures [between 200 and 300 C (390 and 570 F)] and pressures anticipated at depths around 10,000 m, because of the electronics contained in the probe.

While rudimentary development of in situ stress measurements has been accomplished, the down-hole techniques would require significant improvement.

Issues and R&D Requirements

Depth of Hole. The hole depth required for adequate isolation from the biosphere would have to be determined by the geologic medium of interest and by the history and physical condition of that medium. Sedimentary rocks in some instances are considered as potential VDH locations, but only where they are considered to be lower in elevation than circulating groundwater, such as deep basins or hydrologically stable synclines. Crystalline rocks may be the best geologic medium for VDH disposal. Usable hole depth in crystalline rock would be influenced by the depth of ground-water circulation within that rock. Ground-water circulation in weathered granite near the surface in a humid environment will generally be significantly greater than in fresh granite in an arid to semiarid environment.

R&D is required to determine the depth required in various geologic media to minimize the possiblity of significant circulation to ground-water systems. The top of the emplaced waste would still have to be significantly below possible contact with circulating ground water, and would have to be properly plugged and sealed against such contact.

Drilling. The discussion of the present state of development of drilling makes it clear that emplacement of nuclear waste in very deep holes is not possible at this time given that (1) the waste canisters will be 31 to 36 cm (12 to 14 in.) in diameter and (2) the depth required for isolation from the biosphere may be as great as 10,000 m. If it is assumed that these two criteria are valid for the conceptual system, then a number of problems related to drilling would have to be solved to attain emplacement in very deep-holes. The key issue is whether it will be possible to develop the technology to drill to 10,000 m with a bottom hole diameter of approximately 48 cm (19 in.) so that a 36-cm canister could be placed in a mud-filled, fully cased hole.

No increase in the present capability to rotary drill deep wells is expected by the year 2000 without some very significant effort to develop new technology. Currently, there is no industry demand to produce the technology advancement necessary. If sufficient resources were available to advance technology, a 9,000-m hole with a 48-cm (19 in.) diameter might be attainable by the year 2000. Most of the hole would be cased; however, in high strength rocks without gas pressure, the bottom part of the hole might be left uncased. Technology improvements required to reach this depth include:

- New drilling muds capable of operating at temperatures of 370 to 430 C (700 to 800 F)
- High-temperature drill bits, either roller cone or diamond
- New drill pipe, including improved designs and use of improved (high-temperature) steels
- Improved support equipment, such as high-temperature logging and surveying tools and fishing tools
- Improved casing materials (high-temperature steels) and joint design
- High-temperature cements and surface pumps for pumping these cements.

Waste Form and Package Integrity. Criteria currently being proposed for waste forms and packages require total containment within the package for the time period dominated by fission product decay (up to 1000 years). The development of materials to retain their integrity for this period of time at temperatures that would be reached when the ambient rock temperature is 200 to 300 C and under geochemical conditions that would be encountered would require significant effort.

Heat Transfer (Thermomechanical and Thermochemical Factors). Under a normal geothermal gradient of 20 to 30 C/km (60 to 90 F/mi) ambient temperatures in excess of 200 to 300 C (390 to 570 F) are expected at a depth of 10,000 m. The heat released by radioactive decay of the emplaced waste would further increase the temperature of the surrounding rock. The magnitude of this induced temperature increase would be determined by the thermal properties of the rocks and the power output of the waste.

Because of the very large height-to-diameter ratio of the column of radioactive waste, the heat flux from the waste would be mainly in the radial direction, as from an infinite cylinder. The temperature within the heat source itself would be very nearly uniform and would drop very abruptly at the ends. Therefore, from a purely thermal point of view, this geometry would be very favorable. It takes 200,000 years for heat from 5,000-m depths to diffuse to the surface (DOE 1979). The thermally induced effects on the chemical stability and mechanical integrity of the geological formation and upward driving of the ground water would be the most critical issues.

The thermochemical behavior of rocks around a deep hole is not predictable at present. Since controlling factors would be the jointing, fracturing, and fluid content of the rocks, thermomechanical behavior would need to be studied in situ. Heater tests in a variety of rocks at design depths would probaby be necessary to understand the complex response to local high temperature of rock that is water saturated, stressed, and fractured.

Some aspects of thermomechanical behavior of rocks can be studied in the laboratory, however. Since fractured rock is in question, and since characterization of natural fractures is at present impossible, these laboratory studies would involve large samples of rock containing one or more joints, obtained by special sampling techniques. The samples may have to be large (dimensions of several meters). This would require extension of present laboratory testing techniques to test at conditions simulating the in situ environment. The areas where study would be particularly needed include:

- Thermal cracking and other forms of degradation of rock
- Thermoelastic response of intact and jointed rock over a long time frame
- Changes in permeability caused by heating a rock mass
- Two-phase transport of fluids in fractured rock
- Hydraulic fracturing in thermally stressed rock
- Thermal conductivity of hot, saturated thermally stressed rock
- Stress corrosion due to heated ground water in thermally stressed rock.

Emplacement. Most people engaged in drilling for resource exploitation feel that, to prevent collapse, the borehole would need to be kept full of drilling mud at all times. This would include the period during which the canister would be lowered for the waste disposal concept. Getting the waste canister to drop through the drilling mud could be difficult because of the close clearance between the casing and canister. The potential accidental contamination of the drilling mud and lowering cable should a waste package be ruptured would raise numerous questions regarding decontamination techniques and optimum loading methods.

Thus, in addition to a need for substantial research and development on improving the properties of the drilling mud, techniques and equipment would have to be developed to assure lowering and releasing the canisters at depths of 10,000 m and for decontaminating the drilling mud and cable in case of canister failure during this operation.

Isolation from the Biosphere. The principal issue of radioactive waste emplacement in very deep holes is the long-term isolation of the waste from the accessible biosphere (LBL 1979).

In addition to packaging, hole conditions, and hole sealing, a number of other conditions would have to be addressed before long-term isolation from the biosphere could be assured. Several of these involve geotechnical considerations, including:

- An improved understanding of the hydrologic regimes of deep crystalline and sedimentary rock units, including porosity, permeability, and water presence.

- An improved understanding of in situ rock mechanical properties under the high temperature and pressure conditions expected at the required depths and under unusual thermal loading conditions. These properties include strength, deformation, stress state, and permeability.

Additional R&D might be required in the areas of site selecton, site evaluation, and geochemistry (LBL 1979).

Sealing. It is assumed that the sealing system for very deep holes must meet the same time requirement for sealing penetrations used by mined repositories. The primary purpose of the seal would be to inhibit water transport of radionuclides from the waste to shallow ground water or the surface for the specified time period. For integrity to be maintained, the sealing material would have to meet the following requirements.

- Chemical composition - the material must not deteriorate with time or temperature
- Strength and stress-strain properties - the seal must be compatible with the surrounding material, either rock or casing
- Volumetric behavior - volume changes with changes in temperature must be compatible with the enclosing medium.

The seal system would consist not only of plugs within the casing, but also of material to bridge the gap between the casing and surrounding rock. To minimize the possibility of a break in containment, rigorous quality assurance would be required during the placing of several high-quality seals at strategic locations within the borehole.

Therefore, research and development would be needed in two major areas - materials development and emplacement methodology - to ensure permanent isolation. Materials development would include investigating plugging materials, including special cements, as well as compatible casing materials and drilling fluids, which might be incorporated into the sealing system. Because the seal would include the host rock, these investigations should include matching plug materials with the possible rock types. It is conceivable that different plug materials would be required at different points in the same hole.

Emplacement methodology would have to be developed for the particular environment of each hole. Considerations should include all envisioned operations in the expected environment, casing and/or drilling, and fluid removal. Because the emplacement methodology would depend on the type of sealing material, initial studies of sealing material development should precede emplacement methodology development. However, the two investigations would be closely related and there should be close interaction between the two phases. In situ tests should be performed to evaluate plugging materials. Equipment developed should include quality control and quality assurance instrumention.

Logging/Instrumentation. Proper development and operation of a VDH emplacement system would require the collection of reproducible, remotely sensed data on the geologic formation from the bottom of a borehole under high temperature and pressure. Existing logging tools are generally not designed to operate at temperatures exceeding 175 C (350 F).

Remote determinations of water content and flow and in situ stress would need to be addressed to permit preemplacement assessment of down-hole conditions to facilitate VDH system design.

Much of the R&D work under way for logging and instrumentation equipment would be applicable to monitoring equipment for the waste disposal area (DOE 1979).

R&D Costs/Implementation Time

The total cost for research and development for this concept is estimated to be about $730 million (FY 1978 dollars) as derived from DOE (1979). The major portion of this cost, or about $600 million, would be for development of drilling techniques and equipment. The development activity described could be accomplished over a 12 to 15-year period.

Summary

Major uncertainties, shortcomings, and advantages of the concept are summarized below:

- The capability to drill with diameters up to 50 cm holes to a depth of 10,000 meters does not exist and would require a tremendous advance in the state of technology. However, should it be demonstrated that considerably lesser depths, e.g., 3,000 m, are consistent with the concept they can be currently achieved with holes of adequate size.
- The temperature, pressure, and chemical environment at depth would present a potentially very hostile environment for the waste package. Significant advances in materials technology might be required to ensure long lived package design.
- Corrective action, defined as retrievability of emplaced waste, would be unlikely after emplacement.
- The approach is probably not consistent with the philosophy of being able to demonstrate technical conservatism in that design margins are considered small.
- Current methodology does not permit adequate assessment of the at-depth emplacement environment, nor are criteria available for site selection.
- The extreme depth of the concept, and the resulting lengthy path to the biosphere might compensate for many of the drawbacks.

6.1.1.4 Impacts of Construction and Operation (Preemplacement)

During the construction and operation phases, the environmental impacts of the VDH concept would be those common to other drilling and excavation activities. Drilling the hole would raise environmental considerations similar to those for drilling deep holes for oil and gas wells, for uranium exploration and production, and for geothermal and deep rock mining. VDH impacts for these phases would be: the conversion each year of several square kilometers from present land uses to drilling/mining and waste repository activities; disturbance and removal of vegetation; temporary impoundment of water in mucking and settling ponds; accumulation of tailings; alteration of the topography at, and adjacent to, the site; and socioeconomic impacts on housing, schools, and other community services. No special environmental considerations beyond those required for normal drilling would be required.

Health Impacts

Radiological Effects to Man and Environment. As indicated earlier, two different waste forms could be considered for disposal in very deep holes: spent fuel in canisters and encapsulated processed high-level waste. A detailed description of these forms is contained in Bechtel (1979a). Additional assumptions are that both waste forms would have undergone a 10-year decay period prior to emplacement and that secondary TRU wastes would be disposed via a mined geologic repository.

The estimated total occupational whole-body dose from VDH disposal during routine operations would be 4,150 man-rem/yr for the spent fuel waste form and 6,260 man-rem/yr for the HLW form (Table 6.1.1). Of this, 910 man-rem/yr for the spent fuel waste and 920 man-rem/yr for the HLW form can be attributed to the emplacement of waste in the deep hole. The detailed breakdown of doses directly attributable to the VDH concept is presented in Table 6.1.2. Doses attributable to the naturally occuring radioactive materials released during excavation of very deep holes are not included in the estimates.

The estimate of the total nonoccupational whole-body dose from VDH disposal is 380 man-rem/yr for the spent fuel waste form and 180 man-rem/yr for the HLW form (see Table 6.1.1.). Only a very small portion would be contributed by the deep hole -- 7×10^{-6} man-rem/yr and 3×10^{-4} man-rem/yr, respectively, for the spent fuel and HLW forms.

Only nonoccupational doses have been estimated for abnormal conditions and these are presented in Table 6.1.3. Insufficient data are available to allow an estimate of the exposure to occupational personnel during abnormal conditions. It can be only assumed that the exposure would be within regulatory requirements. In this instance, the estimated total

TABLE 6.1.1. Radiological Impact - Routine Operation (Bechtel 1979a)

	Whole Body Dose, man-rem/yr	
	Occupational	Nonoccupational
Spent Fuel		
AFR	1580	320
Packaging and Encapsulation (P/E) Facility	1100	20
Transportation	80	40
Repository (secondary waste)	470	5×10^{-6}
Deep Hole	920	7×10^{-6}
Total	4150	380
HLW		
P/E Facility	4090	90
Transportation	210	90
Repository (secondary waste)	1030	2×10^{-5}
Deep Hole	930	3×10^{-4}
Total	6260	180

TABLE 6.1.2. VDH Concept - Occupational Doses During Normal Operation (Bechtel 1979a)

Operation	Whole Body Dose, man-rem/yr Spent Fuel	HLW
Primary Waste Receiving	170	220
Damaged Canister Receiving/Processing	80	100
Surface Waste Management	40	70
Decommissioning	40	10
Primary Waste Placement	370	320
Interim Confirm. Building	30	30
Support/Overhead	180	170
Total	910	920

whole-body dose would not be applicable because the individual estimates given in Table 6.1.3 cannot be added algebraically. However, note that for both waste forms the potential for the highest exposure would be for a transportation accident, which is not an operation unique to the VDH concept.

Nonradiological Impacts. Nonradiological impacts should be comparable to those of any large construction project and those of industry during operation. Injuries, illnesses, and deaths common to such operations might be expected.

TABLE 6.1.3. Radiological Impact - Abnormal Conditions[a]

Operation	Whole-Body Dose, m rem/event (Nonoccupational)
Spent Fuel	
AFR	2×10^{-3}[b]
P/E Facility	3×10^{-1}
Transportation	1100[c]
Repository (secondary waste)	60[d]
Deep Hole	60
HLW	
P/E Facility	3×10^{-1}
Transportation	1100[c]
Repository (secondary waste)	60[d]
Deep Hole	70

(a) Dose estimates imply consequences of a design basis accident. No probability analysis is included.
(b) Design base accident (DBA) is tornado.
(c) DBA is train wreck, in urban area followed by a fire.
(d) DBA is hoist failure handling secondary waste.

The occupational hazards during normal operations of the waste disposal system would be expected to be no more, and maybe fewer, than the average associated with the various trade/professional workers required to operate the system.

In the case of routine operation nonoccupational hazards, the expected impact would not be detectable.

There are no specific data available to permit a quantitative estimate of the consequences of accidents that may arise. It is expected that abnormal occurrences such as fires, derailments,.transportation accidents, and equipment failures common to industry would occur, but with reduced frequency. Consequently, the occupational impact would be expected to be less than that for industry in general.

Natural System Impacts

Currently available information is so limited that quantitative estimates of the radiological impact on the ecosystem are not available. However, it is expected that, during normal operations, the impact would be minimal, i.e., not greater than that for the mined geologic repository concept. Engineered safety features would be provided to ensure that the disposal system would operate in compliance with regulatory requirements. In addition, location of the waste in holes as deep as 10,000 m would increase the transport path to several kilometers more than that for the mined geologic repository. This would tend to further mitigate the consequences of radioactive waste leak, should it occur, by increasing the transport time.

Microfractures and other openings might develop in the vicinity of the hole because of the stress relief created by drilling or excavation. In addition, small openings might develop within the cement plug and between the plug and the hole wall if the bonding between the two were not adequate. Such channels would provide pathways for contaminated waters to migrate to the biosphere. If the hole were sited below circulating ground water, the primary driving force for migration would likely come from the thermal energy released by the radioactive waste. The travel time to the biosphere would therefore depend on the availability of water, the continuity and apertures of the existing and induced fractures, the time and magnitude of the energy released, geochemical reactions, and the volume and the geometry at the opening over which the energy persists. The lack of data on the presence of water and the properties of fractures in deep rock environments prevents making any estimate of the consequences to the ecosystem.

Nonradiological effects on the ecosystem might impact both water and air quality. Water quality might be affected by the discharge of treated wastewater to the surface water and by rainfall runoff from graded areas, rock piles, and paved areas. Air quality and meteorological changes would come from the generation of fugitive dust and the creation of reflecting surfaces. Air quality would also be affected by emissions from diesel-powered construction and transportation equipment, stack gases, and fugitive dust. The exact discharge quantities and runoff characteristics and the exact amount and type of construction equipment are not available at this time. Parameters such as vehicle miles, surface areas of structures and pavement, soil characteristics, and size of stock piles are also unavailable. For each of these parameters, a qualitative estimate was developed where the water quality effects are based on total land requirement for the facility. The meteorology and air quality impact estimate was based on the number of construction sites, which represent a variety of dust and diesel emissions, and the number of operational emission sources (Bechtel 1979a). The estimates are given in Table 6.1.4.

Socioeconomic Effects

A complete assessment of the socioeconomic impacts of the VDH concept cannot be made at this time because few data are available. In addition, the data that are available can be used only inferentially. These data, which relate to operating employees and community facilities, indicate that impacts would be only moderate.

These inferences are based on a classification scheme where minor, moderate, and major correspond to less than 2,000 employees, between 2,000 and 4,000 employees, and more than 4,000 employees, respectively. For the community facilities two locations is minor, three to ten locations is moderate, and more than ten locations is a major impact.

Aesthetic Effects

As with socioeconomic effects, only minimal data are available for aesthetic effects and these data can be used only inferentially. The available data relate to visual effects only. In this case, the inference is that aesthetic impact would be moderate for both waste forms.

This inference is based on a classification scheme where:

Minor = no permanent structures, facilities, or equipment more than 100 m high

Moderate = one facility with permanent structures, features, or equipment more than 100 m high

Major = more than one facility with permanent structures, facilities, or equipment more than 100 m high.

TABLE 6.1.4. Nonradiological Environmental Impact

Category	Spent Fuel	HLW
Water Quality Facility Area, ha	2400	800
Meteorology and Air Quality, number of construction sites/operational sources	9/42	0/10

Resource Consumption

The consumption of major resources for each case has been estimated from available literature.

Energy. The estimates of energy consumption in the forms of propane, diesel fuel, gasoline, and electricity are presented in Table 6.1.5 for both the spent fuel waste form and HLW (Bechtel 1979a).

Critical Material Other Than Fuel. The estimated consumption of critical resources is presented in Table 6.1.6 (Bechtel 1979a).

Land. The estimated total land that would be required for a 5,000 MTHM/yr waste disposal system is 14,000 ha (35,000 acres)for the spent fuel waste form and 8,000 ha (20,000 acres) for the HLW form. In both cases, the estimated impact would be moderate.

International and Domestic Legal and Institutional Considerations

The international/domestic legal and institutional considerations associated with a VDH repository are expected to be of the same nature as those addressed for a mined geologic repository. (See section 3.3.2 and section 3.5.2)

6.1.1.5 Potential Impacts Over the Long Term (Postemplacement)

The potential for impacts over the long term would relate both to human activities and to natural phenomena. In turn, human activities could be related to the failure of engineered features or human encroachment. Natural phenomena, such as earthquakes and volcanoes, could also degrade the integrity of the waste repository. The heating, rock alteration, or thermomechanical pulsing that could be caused by wastes reaching critical mass are issues common to other geologic disposal alternatives. These aspects would be dependent on the specific rock and site characteristics, waste form, quantity, and spacing and could be evaluated only when these parameters have been defined.

Table 6.1.5. Estimated Energy Consumption

Fuel Type	Spent Fuel	HLW
Propane, m^3	2.3×10^4	1.0×10^7
Diesel, m^3	1.6×10^7	3.4×10^6
Gasoline, m^3	1.6×10^5	1.2×10^5
Electricity, kWh	2.0×10^{10}	5.6×10^{10}

TABLE 6.1.6. Estimated Consumption of Critical Resources

Material	Spent Fuel	HLW
Carbon Steel, MT	3.3×10^6	6.8×10^5
Stainless Steel, MT	8.4×10^4	2.3×10^4
Components		
Chromium, MT	1.4×10^4	4.6×10^3
Nickel, MT	7.5×10^3	2.0×10^3
Tungsten, MT	3.0×10^3	0.5×10^3
Copper, MT	1.3×10^3	1.9×10^3
Lead, MT	1.3×10^4	2.9×10^3
Zinc, MT	1.2×10^3	0.6×10^3
Aluminum, MT	1.3×10^3	1.2×10^3
Water, m^3	2.0×10^8	5.9×10^7
Concrete, m^3	1.9×10^6	1.3×10^6
Lumber, 10^4 m^3	5.6×10^4	3.8×10^4
Clays, 10^6 MT	9.2×10^6	1.5×10^6

Potential Events

The long-term impact of a VDH repository on the ground-water regime would be governed essentially by the nature of the deep ground-water system. Because of the great depth of emplacement and the larger volume of rock available to absorb the energy released by radioactive decay, the deep ground-water system probably would not be appreciably perturbed by the waste itself. If the deep hole were located within a recharge zone or in a zone of lateral movement, the distance to the biosphere along the path of flow might be so long and the velocities so low that isolation might be effectively achieved. Furthermore, the transport of radioactive contaminants by the flowing water would also be greatly retarded by the increased residence times and the increased time for interaction of the contaminant with the host rock.

Engineering Failure of Isolation Mechanism. The principal engineered isolation mechanism for this waste disposal system would be the containment seal. After emplacing the nuclear waste in the deep boreholes, the holes would be sealed to isolate the waste from the biosphere. This isolation would have to be sustained for tens to hundreds of thousands of years for HLW. Not only would it be necessary to seal the borehole itself, but consideration would have to be given to plugging any damage that could have occurred around the hole.

The loss of the integrity of this containment seal might provide a pathway for the waste into the biosphere. The impact on the environment resulting from such a failure could be evaluated only on the basis of site-specific parameters. The lack of specific data prevents a quantitative evaluation. However, it is not expected that resulting impacts would be any greater than those for a mined geologic repository under comparable conditions and might be less due to the longer pathway of smaller diameter than a mine shaft.

Natural Phenomena. Another concern for the VDH concept in the long term would be the susceptibility of the ground-water system to tectonic changes and volcanic action. The very concept of the deep hole is aimed at minimizing such effects by increasing the distance to the biosphere as much as is technically feasible. Placement of the waste disposal site in a tectonically stable region would reduce the probability of such catastrophic events. Site-specific data would be required to quantitatively assess the impact of natural phenomena leading to degradation of the containment.

Inadvertent Human Encroachment. Human intrusions into the VDH repository in the long term could result from drilling, exploration, and excavations. Monitoring, surveillance, and security operations carried out after the repository were closed would provide an increment of safety against such occurrence. However, the physical depth of the VDH would in itself be expected to provide a significant deterrent against human encroachment.

Potential Impacts

The loss of integrity of the waste disposal system as a result of an engineered system failure, natural phenomena, or human encroachment might give rise to environmental consequences by introducing radioactive waste into the biosphere, which would result in radiological health effects. Similarly, ecosystem effects and nonradiological health effects are conceivable.

Radiological Health Effects. It is difficult to predict the nature of future events that would cause a breach of the barriers isolating the nuclear waste from the biosphere. Hence, it is assumed that the system would perform as designed for a prespecified period of thousands of years (Bechtel 1979a). After the period in which the isolation scheme performs as engineered, the barriers would be assumed to be susceptible to breach by:

- Normal degradation, due to expected, naturally evolving events, such as breach by an aquifer with the eventual leaching and migration of the waste
- Abnormal penetration, due to unexpected events, such as drilling or mining of the waste site by man.

The actual scenarios are described in detail in Bechtel (1979a). The radiological impact is expressed in terms of dose per year or dose per event in the case of the abnormal occurrence. The impacts are given in Table 6.1.7.

Ecosystem Effects. An evaluation of the effects on the ecosystem in the long term requires data that are presently unavailable. However, it is not expected that the impact on the ecosystem would be any greater than that for a mined geologic repository, and maybe less, since the radionuclides would be expected to take longer to reach the biosphere.

TABLE 6.1.7. Long-Term Radiological Impact of Primary Waste Barrier Breach

	Waste Type	
	Spent Fuel	HLW
Normal Events (mrem/yr)		
Whole Body	7×10^{-4}	7×10^{-4}
Bone	5×10^{-4}	5×10^{-4}
Abnormal Events (mrem/event)[a]		
Whole Body	Negligible	Negligible
Bone	Negligible	Negligible

(a) Dose is 50-year dose commitment from 1 year intake to the maximum exposed individual.

Nonradiological Health Effects. Although there are no specific data to evaluate the nonradiological health impact, it is expected that these impacts would be comparable to those found in the corresponding industries, e.g., mining, drilling, and excavating.

6.1.1.6 Cost Analysis

All cost estimates are in 1978 dollars based on January 1979 dollar estimates (Bechtel 1979a) less 10 percent.

The estimates are based on preliminary conceptual design data and were developed without the aid of previous cost estimates for this type of facility. Because of the high uncertainties in the cost of rotary drilled holes as large and deep as are called for in this VDH concept, the costs given should be considered only as preliminary estimates.

Capital Costs

On the basis of the waste system description, as presented in Section 6.1.1.2, the estimate of the capital cost for the spent fuel case is approximately $2.3 billion. For the HLW case, a capital cost estimate is $290 million (Bechtel 1979a).

Operating Costs

Operating cost estimates for the spent fuel case have been calculated per year for years 1 through 38 and then for phasedown years 39 and 40. These costs, which include VDH rotary drilling, moving emplacement structures, hole sealing, and receiving facilities operations, would be about $1.7 billion for each year through the 38th year, $1.6 billion for year 39, and $0.8 billion for year 40.

For the HLW case for the same time periods, estimated costs would be $210 million for each year through the 38th year, $200 million for year 39, and $260 million for year 40.

Decommissioning Costs

Total estimated decommissioning cost for the spent fuel case would be $32 million. Total for the HLW case is estimated at $11 million.

6.1.1.7 Safeguards

As noted, the waste types that can be handled in the VDH concept would be limited by volume constraints. Thus, choosing this alternative would require safeguarding two separate disposal flowpaths. The risk of diversion would be strictly a short-term concern, because once the waste had been successfully disposed of in accordance with design, the waste would be considered irretrievable. Physical protection of the sensitive facilities and transportation operations would be the most effective way to deny access to the waste for the short term, as is common to most waste disposal alternatives. For additional discussions of predisposal operations safeguards see Section 4.10.

6.1.2 Rock Melt

6.1.2.1 Concept Summary

The rock melt concept for radioactive waste disposal calls for the direct emplacement of reprocessed liquid or slurry HLW and remote-handled (RH) TRU into underground cavities. After the water has evaporated, the heat from radioactive decay would melt the surrounding rock, eventually dissolving the waste. In time, the waste-rock solution would refreeze, trapping the radioactive material in a relatively insoluble matrix deep underground. The waste and rock should achieve reasonable homogeneity before cooling, with resolidification completed after about 1,000 years. Rock melting should provide high-integrity containment for the radionucles with half lives longer than this period. Spent fuel and secondary wastes (hulls, end fittings, and contact-handled (CH) TRU are not suitable for rock melt disposal unless they could be safely and economically put into a slurry for injection. Otherwise, they would be disposed of using some other form of terrestrial disposal, such as a mined geologic repository.

The waste-rock solidified conglomerate that would ultimately result is expected to be extremely leach resistant, to the extent that it might provide greater long-term containment for the waste isotopes than a mined geologic repository. Because less mining activity would be involved, the cost advantages could be substantial (Bechtel 1979a).

After emplacement, the waste would be considered to be irretrievable, although it could probably be recovered at great expense during the charging or waste addition period while cooling water was still being added. However, the recovery operation would become much more complex and expensive with time as the size of the charge increased (Bechtel 1979a).

There are several technological issues to be resolved and considerable R&D work would be needed before this concept could be implemented. Primary needs would be for better understanding of heat-transfer and phase-change phenomena in rock to establish the stability of the molten matrix and for development of engineering methods for emplacement.

6.1.2.2 System and Facility Description

System Options

The reference concept for rock melt disposal of nuclear waste has been developed from a number of options available at each step from the removal of spent fuel from the reactor to disposal in the rock melting repository.

Various options to be considered are summarized in Figure 6.1.3. The bases for selection of options for the reference concept (those blocked off) are discussed in detail in various documents listed in Appendix M. In addition, a number of options for variations within the concept were considered. These options could improve the concept by changing the cavity construction method or the waste form, or by eliminating cavity cooling (Bechtel 1979a and DOE 1979).

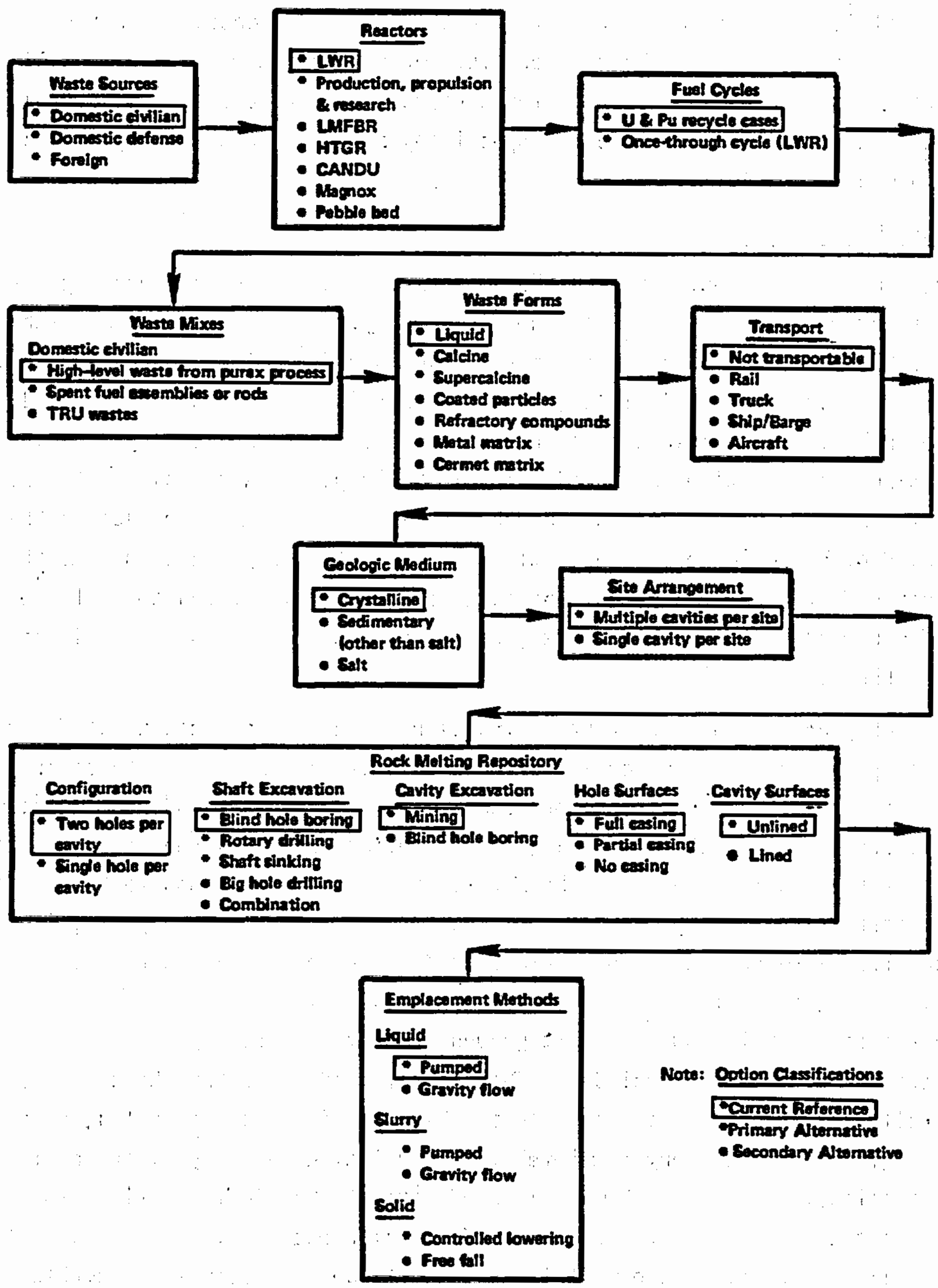


FIGURE 6.1.3. Major Options for Rock Melting Disposal of Nuclear Waste

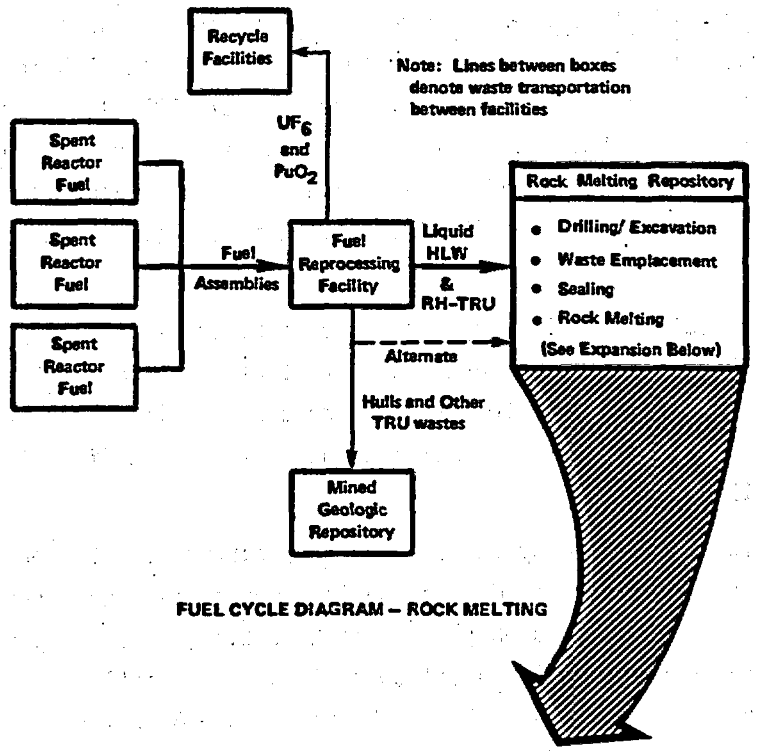
Waste-Type Compatibility

It is assumed for the reference case that only liquid HLW and liquid RH-TRU would be injected into the rock melting cavity. Because of uncertainties associated with emplacement, such as additional criticality concerns, and a sufficient heat generation rate for the volume, spent fuel is not considered suitable for this reference case. Therefore, spent fuel and other wastes that may have low heat generation per unit of volume, such as solid RH-TRU and CH-TRU, are assumed to be sent to a geologic repository. Note that the suitability of spent fuel and other wastes for rock melt disposal may be improved by safely and economically putting them into a slurry form.

Waste-System Description

Basically, rock melting would work in the following manner. In the charging phase, HLW in aqueous solution would be injected into a mined cavity. The heat generated by the radioactive decay of the waste would drive off steam, which would be piped to the surface. When the boil-off rate reached a certain level, liquid transuranic wastes would be added to the charge. Periodically, high-pressure cleaning water would be flushed through the injection piping to minimize contamination and solid particle buildup. This cleaning water would also flow into the waste, providing a coolant to prevent the rock from melting during the waste charging phase. Cooling would be by evaporation or the heat of vaporization. At the surface, the steam driven off from the waste would be condensed and recirculated to cool the charge in the cavity. The closed system would be designed to prevent the release of radioactivity to the environment (Bechtel 1979a).

After about 25 years, when a substantial fraction of the cavity volume was filled, charging would be stopped. After the water was allowed to boil off and the waste to dry, the inlet hole would be sealed. The cavity temperature would rise rapidly and rock melting would begin, with radioactive materials dissolving in the molten rock. As the mass of molten rock grew, its surface area would expand and the rate of conductive heat loss to the surrounding rock would increase. Preliminary calculations indicate that at about 65 years, the rate of conductive heat loss from the melt pool would exceed the rate of heat input from radioactive decay. At this point, the melt would begin to slowly solidify. During the rock melting phase, the heat from the melt would inhibit ground water from entering the area and should prevent the leaching of the radionuclides. This is referred to as the "heat barrier" effect (DOE 1979). Following resolidification, when the heat barrier had dissipated, fission products would have decayed to very low levels. The relative toxicity of the residual radionuclides in the solidified waste-rock matrix is expected to be significantly less on a volumetric basis than that of a typical uranium ore from which nuclear fuel was originally extracted. The final product of the melt is expected to be a relatively insoluble sphere or resolidified silicate rock conglomerate, with a highly leach-resistant matrix, which would be deeply isolated from the biosphere (Bechtel 1979a).



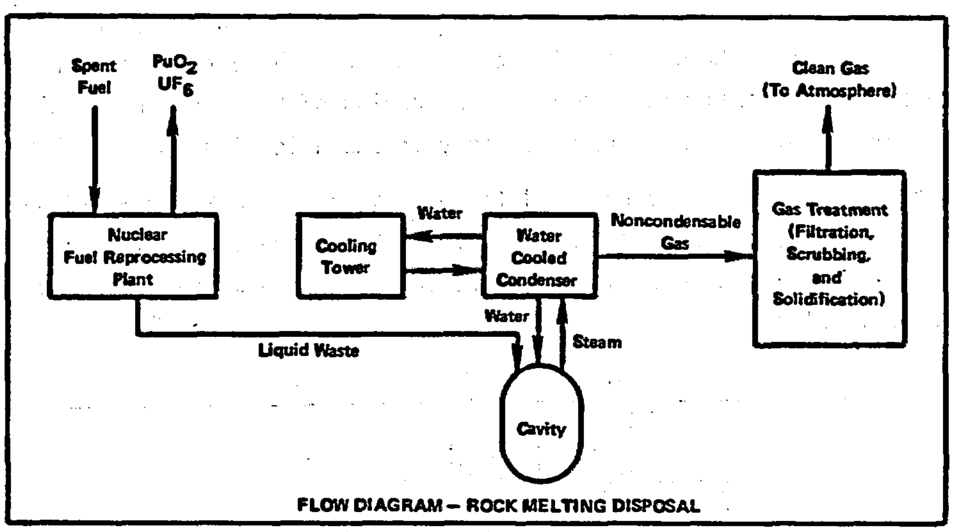


FIGURE 6.1.4. Waste Management System-Rock Melting Disposal

The reference concept design for rock melt disposal was selected through judgment of a "most likely" approach based on available information and data and is not supported by a detailed systems engineering analysis. The fuel cycle and process flow for this concept are shown in Figure 6.1.4. In the reference concept, a repository is designed for disposal of 4 million liters per yr (5,000 MTHM/yr) of high-level liquid waste (HLLW) for 25 years. This requires three 6,000 m^3 (212,000 ft^3) cavities, about 2,000 m (6,560 ft) below the surface on a single site. The three cavities would be located about 2,000 m from each other (Bechtel 1979a).

Predisposal Treatment of the Waste. The reference concept requires a fuel reprocessing plant to recover uranium and plutonium for recycle and to generate HLLW for disposal in the rock melting cavity, as described in Appendix VII of Bechtel (1979a). This plant could be located either on or off site, but the reference concept assumes an on-site location because of restrictions on the transportation of liquid radioactive materials. If solid pellets were produced in the packaging/encapsulation (P/E) facility, an off-site location would be feasible.

Site. The primary factor in selecting a site would be the suitability of the rock formations. Those rocks of greatest interest as potential media for rock melt disposal are composed of silicate minerals. Silicate mixtures are characterized by a melting interval rather than a definite melting point, the melting interval being different for each different set of minerals (DOE 1979).

The melting interval is bounded by the solidus temperature (the temperature at which liquid first forms as the rock is heated) and the liquidus temperature (the temperature above which mineral crystals do not exist stably). In rock melting, these temperatures would depend on parameters such as pressure, chemical composition (especially the amount of water present) and the state of segregation of the rock (see Figure 6.1.5) (Piwinskii 1967, Luth et al. 1964, and Wyllie 1971a). Therefore, the ultimate size of the rock melt cavity would depend on the waste decay heat level and the rock characteristics, including thermal conductivity and thermal diffusivity. Also , the ultimate volume of the molten rock would be influenced by the size of the original mined cavity. The radius of the waste-rock melt pool, as a function of time, for a typical rock melt repository is shown in Figure 6.1.6 (DOE 1979).

The total site area that would be required for a rock melt repository would depend on the number of cavities, the size of the cavities, spacing between the cavities, and surface facility requirements. For this reference concept, the site area would be approximately 4 km^2 (1.5 mi^2) (Bechtel 1979a).

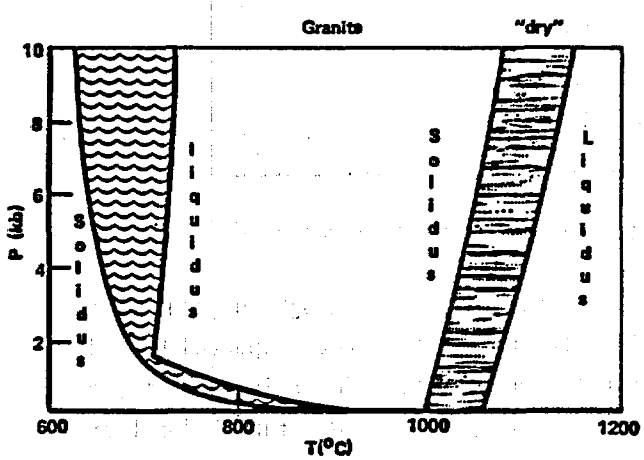


FIGURE 6.1.5. Schematic Illustration of Hydrous and Anhydrous Melting Intervals for an Average Granite

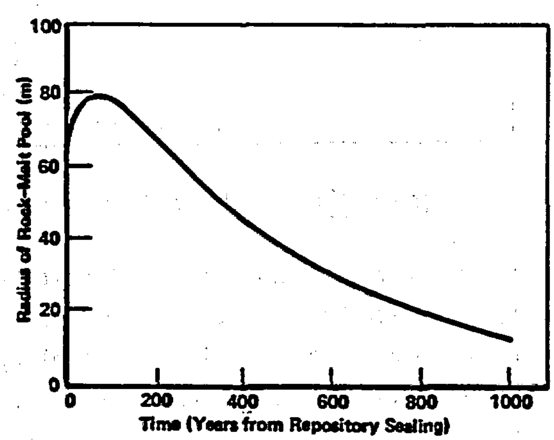


FIGURE 6.1.6. Radius of Waste-Rock Melt Pool Over Time (For Typical Cavity and Waste Loading)

Drilling/Mining System. The reference concept requires two access shafts for each cavity, each 2 m (6.6 ft) in diameter and approximately 2,000 m (6,560 ft) deep. They would be drilled using the blind hole boring method (Cohen et al. 1972). A rotating head with cutters would be turned by electric motors down hole. The entire boring machine would be held fixed in the hole by a hydraulic gripping arrangement. The shafts would be lined with carbon steel casings after drilling (Bechtel 1979a). This method would require men in the shaft to operate the boring machine (DOE 1979).

The cavity would be excavated by conventional mining techniques, although the equipment used would be limited by the access shaft diameter (Bechtel 1979a). Any blasting would be controlled to minimize fracturing of the surrounding rock. The spoil from both drilling and excavating would be hoisted up the access shafts by cable lift for surface disposal (Bechtel 1979a).

Repository Facilities. If the reprocessing plant were located on site, the reprocessing facilities would include a processing/packaging facility. If processing and packaging of wastes for off-site disposal were performed off site, the repository facilities would include a receiving facility similar to that described for the very deep hole concept (Section 6.1.1.1). The following description assumes that the reprocessing facility would be on site.

Four identical stainless steel tanks would be provided for storing HLLW. These tanks would have a combined capacity of about 10^6 liters (2.8×10^5 gal), which equals 3 months' production. The tanks, with the same design as those at the commercial reprocessing plant in Barnwell, South Carolina, would be contained in underground concrete vaults and provided with internal cooling coils and heat exchangers to prevent the waste from boiling (Bechtel 1979a).

An underground pipe system would connect the reprocessing facility to the storage tanks and the three rock melting cavities. The pipe would be double cased and protected by a concrete shielding tunnel. The pipe annulus would contain leak detectors. Heavy concrete and steel confinement buildings over the pipe and cavity shafts would provide for containment, shielding, monitoring, decontamination, maintenance, and decommissioning activities, primarily by remote control (Bechtel 1979a).

There would be four main pipes in the operating shaft to the rock melting cavity:

- A double-wall, stainless steel waste-addition pipe
- A single-wall, stainless steel water-cooling pipe
- A single-wall, stainless steel steam-return pipe
- A stainless steel instrumentation pipe through which monitoring devices would be inserted to measure the temperatures and pressures at various points in the system (Bechtel 1979a).

The confinement buildings over the cavities would also house the equipment and systems needed for filling the cavity and sealing the shaft. Three important process systems would be: (1) the pipe and valve manifold enclosure, (2) the condensing plant, and (3) gas processing equipment. Pipe and valve manifolding would be located in an enclosure near the top of the cavity operating shaft. The cooling water injected into the cavity and the steam from the cavity would be routed through this enclosure. There would be an operating and instrumentation gallery adjacent to the enclosure (Bechtel 1979a). (The HLLW would be charged through a separate underground pipe, mentioned above, that would not go through the confinement building or the pipe and valve manifold enclosure.)

The condensing plant would cool and condense the steam coming out of the cavity and recycle it as cooling water during the waste charging phase. The potentially radioactive primary cooling loop and the nonradioactive, closed-circuit intermediate cooling loop, along with the associated pumps and heat exchangers, would be shop fabricated in modules and designed for rapid remote maintenance. Since the rock would start to melt in a matter of days without cooling, all heat exchanger and pump systems would be designed and constructed with full redundant capacity to ensure constant cooling.

Most of the gaseous elements in spent fuel would be removed during reprocessing at the fuel reprocessing facility. However, some fission product iodine in the liquid wastes could become volatile during the waste charging phase and would be carried out with the steam. This would be trapped by the gas processing equipment and returned with the cooling water to the waste charge or packaged for disposal in a mined geologic repository (Bechtel 1979a).

Auxiliary facilities would support the systems and equipment located inside the confinement building. These would include the water treatment plant, cooling tower, and radwaste treatment (Bechtel 1979a).

<u>Sealing Systems</u>. There would be two principal shaft sealing operations:

1. Sealing of the spare shaft after construction and before waste charging begins.
2. Sealing of the charging shaft after completion of waste filling but before rock melting begins.

The NRC's Information Base for Waste Repository Design (NRC 1979) provides recommendations for sealing conventional boreholes and shafts. Though this information base may not be particularly applicable to the rock melt concept, it states that removal of the steel casing is essential for long-term performance of the seal. The seal must be bonded directly to the geological strata for maximum strength. Expansive concretes make the best seals under current technology and do so at an acceptable cost. However, it is not certain that these seals, whether cement, chemical, or other material, will successfully resist deterioration over a period of 1,000 years on the basis of current penetration sealing technology. Seal failure must be assumed even for seals placed under carefully controlled conditions using state-of-the-art technology and materials. Further development of sealing technology would, therefore, be required (DOE 1979).

Postemplacement sealing of the pipes within the shaft, the shaft itself, and the pipes and valve gallery in the confinement building would be a more complex problem. This is because of the limited time, the high temperatures involved, and the radioactivity levels in the system. Considerable technology in this area has yet to be developed, as discussed in the following section.

Retrievability/Recoverability. Wastes disposed of by this concept would possibly be retrievable for a short period. Prior to melting, most of the liquid or slurry could be removed. After the melt has begun, well techniques for the molten rock-waste mixture might be possible. However this is unproven and would likely be an expensive and difficult process. Postclosure recovery of the solidified waste form would require extensive mining and excavation of large quantities of hot and molten rock containing waste.

6.1.2.3 Status of Technical Development and R&D Needs

Present State of Development

Substantial fundamental and applied research would be required for continued development of the rock melting disposal concept. This method is in the conceptual stage and no experimental work has been undertaken to support its feasibility.

Rock Melting Process. Generally, rocks are multiphase mixtures of a number of minerals characterized by a melting interval, as noted earlier. Because any two samples of a particular type of rock will have slightly different mineral compositions, they will also have slightly different melting intervals. As we have seen, the boundaries of these intervals (liquidus and solidus temperatures) depend on several parameters.

If the composition of the rock in which a waste repository were to be located has been well characterized, the melting properties of that rock could be predicted with some precision, and if the thermal conductivity, thermal diffusivity, and the heat of fusion of the rock were also known, the melting "history" of the HLW/rock melting phase could be predicted.

Clearly, it would be prudent to experimentally verify such predictions by means of prototype experiments; however, it should not be necessary to carry out an extensive series of such experiments to verify the current predictive capability for estimating the rate of rock melting and the total amount of rock melted for a particular set of waste repository conditions.

Effects of Heat on Rock Properties. The properties of rock subjected to high thermal gradients would be important inputs to determining the condition of the rock enclosing the molten waste-rock matrix. While the radius of this molten zone should be small compared with the extent of the geologic formation in which the repository would be sited, the zone's properties would have to be known so that an appropriate structural and safety analyses could be carried out.

The inner edge of this zone would be defined by the maximum radius of rock that had been heated to liquid formation. The outer radius of the zone could be roughly characterized as that location beyond which the rock had not been measurably affected by heat from the HLW.

The heat effects in the peripheral edges of the zone would be similar to effects found in a mined repository.

Transport of Radionuclides in Rock Melting. Under normal operating conditions, the casing in the emplacement well should prevent contact of radioactive waste with any aquifers that would overlie the disposal cavity. However, during waste charging, it is conceivable that some radioactivity could migrate out of the cavity into the surrounding rock. But, if the cavity were maintained approximately at atmospheric pressure, the tendency of water under hydrostatic pressure to flow into the cavity should minimize the importance of this transport mechanism.

During the rock melting phase, transport of radionuclides out of the waste-rock mixture would presumably be inhibited, because no water would be present in the melt and a portion of the surrounding zone of heated rock (Taylor 1977). (This is the "heat barrier" effect referred to earlier.) However, the radionuclide leaching capabilities of the high-pressure and high-temperature water vapor existing in this region would have to be characterized.

Finally, after the waste-rock matrix had cooled and solidified, it must be assumed that water would reenter the matrix and leach at least some of the radionuclides out of the matrix volume. Leaching potential at elevated pressure and temperature would have to be determined. As the radionuclides were transported to the relatively cool rock away from the repository, existing data on radionuclide transport in rock should be applicable (Klett 1974, Burkholder et al. 1977, de Marsily et al. 1977, Pines 1978, EPA 1978). It is possible that leaching data on other waste forms could also be useful (Brownell et al. 1974, Ralkova and Saidl 1967, Schneider 1971b, Mendel and McElroy 1972, Lynch 1975, and Bell 1971).

Effect of Superheated Water on Glasses in Rock Melting. Data from recent investigations of the devitrification of glass by water at high pressure and temperature (McCarthy et al. 1978 and McCarthy 1977) could be useful in determining the availability of radionuclides to water from vitrified rock present in the resolidified waste-rock matrix. However, the applicability of the conditions under which these data were obtained to the rock melt concept would have to be established.

Safety Studies: Disposal of HLW with Rock Melting. During the cavity charging portion of the presealing phase, HLW in such forms as solutions or slurries would be directly introduced into the repository cavity. The various operations that would be involved in carrying out this phase of the process are not as unique as the postsealing phase. Consequently, the probabilities for the release of radioactivity to the environment can be estimated for each step of this phase. This can be done both for normal operation and for assorted accident scenarios. In general, sufficient data exist to prepare a risk analysis for this phase of the rock melt concept.

After cooling of the waste-rock matrix to the point where water could contact the waste, it may be assumed for purposes of modeling that the waste dissolves, and transport through the surrounding rock is initiated. Calculations for risk analysis of this postsealing phase are identical with those used for the risk analysis of other geologic waste disposal concepts with the exception of possible bulk migration of the molten mass during the interim phase between cavity sealing and solidification.

Ground Water Migration and Rock Melting. While a molten or high-temperature rock mass would disrupt natural patterns of water movement in the vicinity of a repository, the relative effect would diminish with distance, until, at some point, the repository would have no appreciable effect on water transport of radioactive materials. Presumably, if the hydrology of the repository area were well characterized, its effects could be modeled by treating it as a roughly spherical barrier with a radius that shrinks as the waste-rock matrix cools. Preliminary work on a laboratory scale and at atmospheric pressure indicates that this "thermal barrier" effect (Taylor 1977) could be demonstrated experimentally; however, additional work that more closely simulates conditions expected at the repository depth would be required.

Technological Issues

The technological issues that would require resolution before initiation of the rock melting concept can be summarized as follows:

- The necessary geological information cannot be predicted with present knowledge.
- Empirical data on the waste/rock interaction and characteristics are lacking.
- No technical or engineering work design of the required facilities has been attempted.

It is not possible at this time to produce a design for the rock melt repository because the necessary information is lacking. Data on the form and properties of the waste to be charged into the cavity, the charging methodology, the properties of the host rock, and many technical aspects of the shaft sinking method and cavity construction technique would have to be resolved. For many of these operations, work could not begin until fundamental waste/rock properties are better known.

In addition, the concept would require operations and process activities that do not readily lend themselves to the same degree of conservatism normally utilized in the nuclear field. Discussed below are several areas that would require further scientific or technical work.

Cavity Design and Construction. The greatest problem might lie in the construction of the cavity. Although, it is within the bounds of current technology to lower men and equipment through a 2-m-diameter shaft and construct the required cavity, such operations are difficult and time consuming. Methods for lining the cavity may have to be developed. Furthermore, it is practically impossible to construct the cavity without cracking the surrounding rock. Since it may be necessary to maintain the waste inside the cavity for some years before rock melting is permitted to begin, it would be necessary to ensure that waste does not escape into the cracks and ultimately into ground water. It may be difficult to assure the necessary leaktightness of the mined out cavity. All of these areas would require technical resolution before construction could begin.

Cavity Charging. Cavity charging methods would depend on many variables including: the radioactivity of the charge; whether the charge were liquid or slurry; whether charging were batch or continuous; and whether charging were a long-term or short-term operation. The methodology for charging has not been defined or optimized. Considering the heat of the waste, the depth of the cavity, and possible corrosion and material plate-out, considerable technical effort would be required in this area.

In addition, the effect of a 2,000-m-long steam line on cavity charging would have to be determined. A vertical pipe of this length would act as a distillation column. Also, the engineering required to construct such a pipe (i.e., the number and type of expansion joints, effect of bends, etc.) has not been performed.

Shaft Sealing. There would be two phases of shaft sealing: sealing after construction but before waste charging starts and sealing after the waste is emplaced but before rock melting begins.

Sealing after construction would be the easier of the two operations because there would be sufficient time to check the work. However, sealing before rock melting begins would have to be done fairly quickly and in a potentially contaminated environment. Radioactive contamination and possible residual steam venting would present substantial problems in trying to seal the shaft after charging. Because of the number of pipes connecting the cavity to the surface, this operation would require considerable expertise. Both the materials and methods required would need further study and experimentation.

Volatile Fission Products. The quantities and behavior of the potentially volatile fission products would have to be determined. Nuclides in this category include ^{103}Ru and ^{106}Ru. Equipment would have to be designed to trap and remove these products from the waste stream or to return them in the coolant back to the cavity. Alternatively, they might be returned to the processing facility. There might also be a liquid and solid carryover from the steam, which would contaminate the condenser as well as increase the hazard from any potential leak. Practical technical considerations in this area would have to be examined before this concept could ever be considered viable. There is also a potential problem with tritium being carried with the steam.

Criticality Potential. Because 99.5 percent of the uranium and plutonium would have been separated from the spent fuel during reprocessing, the potential for criticality in the HLW is small. If experimental and modeling results indicated that criticality might be attained at some point in one of the rock melt concept scenarios, and if the results of such an excursion were undesirable from either an engineering or a safety standpoint, additional work would have to be carried out to develop methods of mitigation, possibly involving the addition of a high neutron cross section "poison" to the HLW as it is emplaced in the repository. It would be necessary for the "poison" to remain dispersed in the proper place upon cooling.

Fracturing During Cooling. During melting, the waste-rock mass would be expected to expand about 13 percent. During subsequent cooling and contraction, fracturing would have to be expected in the rock zone that surrounds the molten area. Further work would be required to establish that the rock melting concept could provide containment of the waste charge under uplift and subsidence conditions.

Chemical and Physical Effects on Surrounding Rock During Rock Melting. While the rock melting process can be described with some precision (Piwinskii 1967, Luth et al. 1964, Wyllie 1971a, and Wyllie 1971b), the effect of a large thermal gradient on various types of rock has apparently not been similarly investigated (Executive Office of the President 1978). Although in some rocks, the predicted thermal effects of a molten mass of HLW/rock extend over relatively short distances, the extreme thermal gradient would clearly produce chemical and physical effects in the rock (Jenks 1977, National Academy of Sciences 1978). These effects would have to be characterized so that the rock mechanics of rock melt disposal could be adequately modeled and any possible intermediate or long-range effects identified and characterized. It would be necessary to carry out measurements over a range of pressures up to the maximum contemplated lithostatic pressure for a waste disposal cavity.

Interaction of HLW with Rock. At the present time, it is not clear whether the possible chemical reactions between the HLW solution and the rock cavity walls are important to the rock melt concept. However, it is clearly desirable to know how and to what extent such reactions take place, and to predict what the ultimate effect of 25 years of waste solution addition would be. With that information, potential problems could be identified, and mitigating measures could be designed and tested.

After addition of HLW to the cavity were stopped and rock melting begun, it is not known how rapidly and completely the HLW would mix with the molten rock. Because relatively complete mixing of the HLW with the rock appears desirable (to ensure complete dissolution of the HLW in the rock and subsequent immobilization upon resolidification of the matrix), it might be necessary to design the HLW rock melt disposal facility to minimize the viscosity of the molten rock.

Properties of Resolidified Waste-Rock Matrix. Even if it is assumed that the HLW is completely mixed with the molten rock, it is not known whether some of the radioactive species in the HLW might segregate during the long cooling process to form relatively concentrated (and possibly, relatively soluble) inclusions in the resolidified waste-rock matrix (Hess 1960). It is possible that the addition of certain chemicals (at the time that HLW is emplaced) could prevent such segregation, decrease the solubility of some or all of the long-lived radionuclides, or both.

R&D Requirements

Resolving these many uncertainties would require an extensive R&D program, such as that described below.

Data Base Development. Development of an adequate data base would require the conceptual design of one or more rock melt respositories. From these design bases, significant engineering features and critical geologic parameters could be identified. Similarly, the relevant properties of the geologic media would have to be understood in the context of the rock melt concept. Also, properties of materials in the waste handling systems would have to be identified and evaluated to determine the ability of these materials to function in hostile environments.

Laboratory-Scale Studies. To develop an understanding of rock melt mechanisms, extensive scale studies would need to be conducted. Specific areas of study should include:

- Heat transfer and phase-change phenomena for various geologic media
- Waste/rock interactions, particularly at elevated temperatures
- Properties of the resolidified waste-rock matrix
- Properties of engineering materials and their ability to function in the predicted environments
- Studies of actual small scale rock melt systems in laboratory hot cells
- Studies on the potential effects of criticality accidents.

Model Development. Better understanding of rock melt interactions could be gained by applying the data base to development of a predictive model covering heat transfer and related phenomena. The model could then be used for sensitivity analyses to determine the relative importance of various parameters and where research and development effort might best be applied.

Site Selection Methodology. From the systems modeling and other research tasks, it would be possible to identify those technological factors that would have to be considered in site selection. When site selection factors had been identified and evaluated, an optimal site profile could be determined to guide the selection process. Currently there is no methodology for locating a site.

Instrument Monitoring Techniques. Instrumentation for monitoring site selection and operational and postoperational phases of rock melt disposal would have to be identified and techniques for its use developed.

Thermal Analysis and Rock Mechanics. The effects of the melting cycle on the integrity of geologic formations would need to be thoroughly studied. Such effects as thermal expansion and contraction, phase change, and hydrologic change before and after emplacement would have to be assessed.

Pilot-Plant Studies. Laboratory and modeling studies should be complemented by a small-scale pilot-plant study involving actual emplacement of nuclear waste in rock. Such a study would be necessary to validate predictive methods and to assure that no vital factors had been overlooked prior to full-scale implementation of the concept.

Implementation Time and Estimated R&D Costs

In view of the significant technical uncertainties remaining, it is not possible to predict a cost estimate of the required R&D to implement this concept, nor the amount of time it would take.

Summary

Major uncertainties, shortcomings, and advantages of the concept are summarized below:

- There is not a multiplicity of engineered barriers inherent to the concept.
- The temperature, chemistry, and other characteristics of the molten waste-rock mixture are not considered consistent with technical conservatism.
- The required characteristics of a site are not known, and criteria for selection are considered extremely difficult to derive.
- The concept cannot be implemented in a step-wise, technically conservative manner due to the scale required for demonstration.
- Performance assessment capability is perhaps most distant for this concept than for any other.
- Retrievability of the waste is considered to be unlikely, so that corrective action cannot be accomplished.
- The time required for monitoring prior to full solidification (defined as the operational period of up to 1,000 years for this concept) exceeds the likely acceptable life for institutional controls.
- The primary postulated advantage relates to the possibility that the solidified waste form might be more stable than other possible forms.
- Lower mining requirements compared to a mined geologic repository may be a secondary advantage.

6.1.2.4 Impacts of Construction and Operation (Preemplacement)

Potential environmental impacts of a rock melt repository would be similar in many respects to those of a mined geologic repository. Both would require surface and subsurface activities that lead to environmental impacts. This impact analysis focuses on unique aspects of the rock melt concept, and refers to discussions on mined geologic emplacement in Section 5.4 as appropriate.

Health Impacts

Health studies related to the rock melt concept for the disposal of HLW can be divided into two phases: the presealing phase, which includes waste transportation and active operation of the waste disposal facility, and the postsealing phase, which includes the melting and resolidification of the HLW/ rock matrix and its long-term effects. In the following discussion, radiological and nonradiological concerns for the first phase are covered separately.

Radiological Impacts. During presealing operations, waste in solution or slurry form would be introduced directly into the repository cavity. Various operations in this charging phase could lead to release of radioactive material into the environment.

Under normal operating conditions, the casing in the emplacement well should prevent contact of radioactive waste with any aquifers that would overlie the disposal cavity. During waste charging, however, it would be possible that some radioactivity could migrate out of the cavity and into the surrounding rock. This possibility would be reduced if the cavity were maintained approximately at atmospheric pressure. Under these conditions, the tendency of water under hydrostatic pressure to flow into the cavity would minimize the importance of this transport mechanism. Nevertheless, it would be possible for radioactive material to reach man through such migration into the surrounding rock and onto the biosphere.

Operational impacts would vary somewhat, depending on which version of the rock melting concept is considered. If liquid HLW were emplaced directly into a cavity from the processing facility, there would be no impacts due to transportation of the waste. If solid waste were slurried into the repository, impacts of waste transportation from the reprocessing plant to the repository would have to be considered. However, such transportation would have no different environmental effects than would the shipping of such wastes to any other type of repository.

Treatment of HLLW prior to emplacement might be required to enhance the compatibility of the liquid with the rock in which the cavity would be located. This additional treatment step would increase the probability of occupational and population exposures to radiation. Handling and treatment of solidified HLW would also increase the probability of radiation exposure; risk analysis would take into account the details of the required handling and treatment procedures.

A summary of potential radiological health impacts was prepared for the rock melting concept (Bechtel 1979a). This study projected the short-term occupational impacts for a single rock melting cavity, which are presented in Table 6.1.8. For a 5,000 MTHM/yr throughput, it is estimated that three rock melting cavities would be required and that the impacts would be linear (Bechtel 1979a). Occupational impacts prior to the waste reaching the repository, nonoccupational impacts, and impacts from abnormal conditions were also postulated in this study. For this analysis, the consequence of impacts under abnormal conditions was found to be comparable to, or slightly less than, those of the other options. This study, however, did not include any probability analysis and consequently total radiological impacts under abnormal conditions have not been quantitatively determined.

Nonradiological Impacts. The underground portion of rock melt repositories would probably be constructed using conventional mining and drilling techniques. Health impacts would be those typical of any analogous construction project, and would be somewhat dependent on the method chosen (whether the cavity were created by mining, underreaming, explosive springing, etc.).

TABLE 6.1.8. Occupational Dose Estimate During Normal Operation At a Single Rock Melting Cavity

Process Unit	Whole-Body Dose, man-rem/yr
Valve Gallery	120
Offgas Recovery	110
Maintenance	50
Decommissioning	30
Support/Overhead	40
Total	350

Impacts from surface construction would be typical of those associated with the construction of any chemical processing plant. Also, impacts similar to those for the mined geologic repository and discussed in Section 5.4 would be expected for this option.

Natural System Impacts

The effects of rock melting on ground-water migration and transport of radioactivity in the surrounding rock and the possible modeling of these effects are discussed in Section 6.1.2.3. This analysis suggests that heat from the wastes should not affect the thermal regime near the surface.

The principal impacts on natural systems associated with HLW disposal are considered to be those normally encountered in underground drilling and construction activities. Construction impacts could be estimated relative to those from conventional repositories on the basis of the amount of excavation required.

Such topics as disposal of mined spoil, emissions from machinery used in construction, and prevention of water pollution from mud pit overflow could best be analyzed for a specific site. General impacts, however, would be similar to those discussed in Section 5.4.

Because of the lack of formal studies, the effects of the melting cycle on the integrity of the geologic formation would need to be thoroughly studied. Effects such as thermal expansion and contraction, phase change, and hydrologic change during pre- and postemplacement environments would have to be assessed. These effects could be significant, but present data are insufficient to draw meaningful conclusions.

Socioeconomic Effects

Overall, the potential socioeconomic impact of a rock melt repository is rated as minor (Bechtel 1979a). This conclusion is reached, in part, because only a moderate sized work force (between 2,000 and 3,000 people) would be required for successful operation. Land requirements would be less than for any of the other disposal alternatives studied (Bechtel

1979a). In addition, with colocation of three rock melting cavities and three reprocessing facilities at each site, only two facility site locations would be required. The resultant fiscal impact on community facilities would therefore be relatively small.

Although rock melt might have the least socioeconomic impact of any of the alternatives, it is impossible to fully address the nature and extent of impacts at the generic level. This is particularly true when analyzing the socioeconomic impact of construction activity--a detailed estimate of the construction work force has not been completed. Nevertheless, it is reasonable to conclude that socioeconomic impacts would be similar to, and generally slightly less than, those described in Section 5.6 for the mined geologic repository. A cautioning note, however, is that colocation of facilities could lead to a concentration of impacts.

Aesthetic Effects

Facilities associated with a rock melt repository would have an aesthetic impact. The extent of this impact would depend on characteristics at the site and would reflect the fact that optimal engineering design would be necessary for different forms of HLW. Facility design would be a function of the physical and chemical form of the HLW.

The extent of surface construction would depend on the rock melting concept version for which the repository was being designed; where HLW solutions were being directly emplaced, the entire reprocessing plant would be located close to the repository. Where waste slurries were emplaced, only a relatively simple surface installation would be required to condense steam, add makeup water, provide for slurry mixing, etc. Aesthetic impacts would reflect final facility design, with larger facilities generally having greater impacts. Overall, aesthetic impacts would be similar to those described for a mined geologic repository, as presented in Section 5.6, with minor exceptions.

Facilities that would be different from those in the mined geologic repository include the type of cooling towers and tall drill rigs used in excavating the rock cavities. In addition, although a 100-m-high stack would be required for a processing facility, its location on the same site as the repository would reduce overall aesthetic impacts. Other aesthetic impacts, such as noise and odor, have not been identified as a problem with rock melt.

Resource Consumption

Energy would be required to construct and operate a rock melt disposal system. Initially, energy would be consumed in transportation and construction activities. In the operational phase, waste preparation, transportation, and emplacement activities would consume energy. Quantitative estimates of energy consumption for the construction and 40 year operation of a 5,000 MTHM/yr system have been prepared (Bechtel 1979a). These estimates are presented in Table 6.1.9.

Consumption of other critical materials has not been identified as an important factor in evaluating the merits of the rock melt concept. Drilling activities, as well as construction of the facilities, would require steel, cement, and other construction materials typically associated with a major facility. Estimates of these requirements are presented

TABLE 6.1.9. Estimated Energy Consumption (Bechtel 1979a)

Propane, m^3	1.0×10^6
Diesel, m^3	1.5×10^6
Gasoline, m^3	1.5×10^5
Electricity, kWh	5.7×10^{10}

in Table 6.1.10 (Bechtel 1979a). No scarce or otherwise critical material has been identified as being important for this option.

As noted, the reference concept calls for each rock melting repository site to support three 6,000 m^3 cavities about 2,000 m below the surface (Bechtel 1979a). Each site would be able to accommodate waste from 5,000 MTHM/yr for 25 years. Construction of these facilities would disturb 1,100 hectares (2,720 acres) of land and would require a restricted land area of 4,000 hectares (9,880 acres) (Bechtel 1979a). Most of the land disturbed would be required for processing, encapsulation, and other surface facilities.

International and Domestic Legal and Institutional Considerations

The rock melting concept would have relatively few international implications because waste transportation activities would occur in the U.S. and emplacement would be achieved well out of range of the biosphere. There are, however, important domestic legal and institutional considerations that would need to be resolved. For example, as noted in Section 6.1.2.2, retrieval of wastes, even before emplacement activities were complete, would be very difficult. The hot nature of the wastes and the type of waste packaging that would be employed would influence the ease with which the waste material could be withdrawn. Retrieval after the cavity was sealed and the waste was in a molten form would be impossible. Legal and regulatory implications of these restrictions on retrieval would have to be resolved.

Selection of the rock melting concept would also affect certain decisions regarding interim storage. If waste from the uranium-only recycle, or the uranium and plutonium recycle were stored, it would be necessary to specify the form of waste storage that would have the least environmental and economic impact. Although it is possible that the waste

TABLE 6.1.10. Estimated Material Consumption (Metric Tons)

Carbon steel	300,000
Stainless steel	24,000
Components	
Chromium	4,800
Nickel	2,200
Tungsten	--
Copper	1,900
Lead	2,900
Zinc	600
Aluminum	900

would be stored as a liquid, it is more probable that it would be solidified (calcined or vitrified) if an extended storage period were envisaged.

6.1.2.5 Potential Impacts Over the Long Term (Postemplacement)

Although repository-related human activity would be minimal once emplacement and repository decommission activities were complete, impacts could occur because of the possible mobility of the molten waste material in the geologic environment. Potential events and impacts are described below.

Potential Events

For risk analysis purposes, the postemplacement phase of the concept is treated in a manner similar to other geologic disposal alternatives (see Section 5.6). As noted earlier, after the waste-rock matrix cooled to the point where liquid water could contact the waste, it is assumed that the waste would dissolve, and transport through the surrounding rock would be initiated. Clearly, the degree of risk calculated on this basis would be strongly site specific, and would depend on factors such as the depth of the repository, presence and location of aquifers, water quality, and sorptive properties of the rock.

Possible pretreatment of the wastes to minimize potential adverse postemplacement effects would depend on the waste form as well as the geologic media characteristics.

Potential Impacts

Basically, the environmental considerations involved in evaluating the long-term impact of rock melting are how much of the radioactivity in the repository would reach the biosphere, when it would get there, and what its effects would be.

The heat barrier effect is discussed in Section 6.1.2.3. Following total resolidification (1000 years), when the heat barrier no longer existed, most fission products would have decayed to innocuous levels. The toxicity of the residual radionuclides in the resolidified waste-rock matrix at that time should be significantly less than that of a typical uranium ore body from which the nuclear fuel was originally extracted.

Mixing of the HLW with the molten rock, as well as the physical and chemical properties of the cooled and resolidified waste-rock matrix, would determine the rate at which radioactive species could be leached and transported by ground water. It might be possible to design some mitigating measures to significantly retard leaching rates of all or some of the radioactive species present.

It is possible that the heat barrier effect would retard the start of effective leaching of radioactivity until radioactive decay had essentially eliminated the fission products as significant health hazards; thus, it might be necessary to consider only the TRU products.

Transportation of radioactivity by ground water would have to be evaluated on a site-specific basis, although different scenarios could be postulated to obtain order-of-magnitude estimates of the time required for radiation to appear in the biosphere and of the concentrations of radioactive species that would be present in the water. In modeling the radioactivity transport, movement of water would be considered as taking place both through permeable rock and by means of joints and cracks in low-permeability rock (Heckman 1978). The impacts of a ground-water breach of a rock melt repository are expected to be similar to those that would result if a mined geologic repository were breached by ground water (Bechtel 1979a).

6.1.2.6 Cost Analysis

Cost estimates for the rock melt concept do not have the benefit of a reference conceptual design, nor of previous cost estimates for similar types of facilities. Therefore, these cost estimates are only approximate. They are based on the reference concept disposal of HLW from 5,000 MTHM/yr, for 25 years, requiring three cavities.

All cost estimates are in 1978 dollars based on January 1979 dollar estimates (Bechtel 1979a) less 10 percent.

Capital Costs

The capital cost of a rock melt repository with an operating lifetime of 25 years is estimated at $560 million.

Operating Costs

An allowance of 2 percent of the capital cost is assumed for the annual operating cost, which comes to $11 million a year.

Decommissioning Costs

The total decommissioning cost for the three-cavity rock melting concept is estimated at $21 million. In this estimate, final shaft sealing is treated as a decommissioning cost with an allowance of $2 million per cavity.

6.1.2.7 Safeguard Requirements

Because of the restrictions concerning the transportation of radioactive liquids, the fuel reprocessing plant would have to be colocated with the rock melt repository. Therefore, accessibility to sensitive materials would be extremely limited with liquid emplacement. If the waste were to be placed in a solid form (e.g., pellets), which could be emplaced in the subsurface cavity as a slurry, the fuel reprocessing plant could be located off site but transportation related safeguards would then be required. The subsurface cavity would increase the difficulty of diversion and the liquid or slurry waste form would complicate the transportation and handling problems for potential diversion. However unlikely, retrieval by drilling and pumping is possible. This would eventually need to be considered for rock melt repository safeguards. Material accountability would be enhanced by ease of sampling and measurement, but gross accountability (i.e., gallons vs. canisters) would be slightly more difficult than for the mined geologic repository concept. For additional discussion of predisposal operation safeguards see Section 4.10.

6.1.3 Island Disposal

6.1.3.1 Concept Summary

Island-based disposal would involve the emplacement of wastes within deep, stable, geological formations, much as in the conventional mined geologic disposal concept discussed in Chapter 5 with an over-water transportation route added. The island would provide port facilities, access terminals, and a remote repository location with possibly advantageous hydrogeological conditions. An island disposal facility could also provide an international repository if the necessary agreements could be obtained.

The island disposal concept has been referred to as an "alternate geologic approach" (Deutch 1978) in which the geology (i.e., rock, sediments) provides the primary barrier between the nuclear wastes and the biosphere and the ocean may provide an additional barrier, depending on the repository location and the hydrological system existing on the island.

The status of the concept is uncertain. The U. S. Department of Energy Task Force Draft Report (Deutch 1978) stated that "The Department of Energy has no program to actively investigate the concept. Suggestions for assessment of the concept have been made from time to time by groups considering international aspects of radioactive waste repositories. However, a consensus for the need of such repositories has not developed."

On the other hand, the sixth report of the U. K. Royal Commission on Environmental Pollution (Flowers 1976) referred to island locations when considering hard rock sites for a geologic facility. In this report, it was stated that "A deep disposal facility on a small uninhabited island would be particularly advantageous if one were chosen which was separated hydrogeologically from the mainland. Any leakage of radioactivity into the island's ground water would be easily detected and in that event the dilution of seawater would provide a further line of defense."

No detailed studies of the island concept are currently available; therefore, its basic elements are based on simplified modification and adaptations of conventional mined geologic disposal as discussed in Chapter 5. Since the geology of most islands is crystalline rock, it is the assumed disposal formation. Elements of other schemes (e.g., subseabed disposal, Section 6.1.4) have been incorporated and/or referenced where appropriate. If more detailed assessments are required in the future, conceptual design studies would have to be performed to provide a reliable basis for analysis.

6.1.3.2 System and Facility Description

System Options

The reference concept for the initial island disposal of nuclear waste has been developed from a number of options available at each step from the reactor to disposal in the island geology.

Various options to be considered for island disposal are summarized in Figure 6.1.7, with options for the reference concept designated. Details on the bases for selecting reference concept options are covered in various documents listed in Appendix M.

Because system options for island waste disposal beginning with the reactor and including steps up to the transportation requirements are similar to those for mined geologic repositories, the options selected for the reference design are similar for the two concepts. From that point on, the selected options are based on current program documentation.

Waste-Type Compatibility

An island repository could handle all wastes from the uranium and plutonium recycle case, and from the once-through cycle.

Waste-System Description

The reference island repository design is based on the concept discussed in Section 6.1.3.1 and the waste disposal cycle options identified above. The fuel cycle and process flow for the reference concept are shown in Figure 6.1.8. The reference system assumes the transport of all spent fuel, HLW and transuranic wastes to the island sites.

The waste forms and emplacement concept of canistered waste for island disposal would be the same as those for conventional mined geologic disposal discussed in Chapter 5.

Predisposal Treatment and Packaging. The predisposal treatment of waste for the island disposal concept would be identical in most respects to the predisposal treatment of waste for mined geologic repositories. Chapter 4 discusses the predisposal systems for both spent fuel and HLW common to all of the disposal concept alternatives.

Geologic Environments. The geohydrologic regime of an island, as diagrammed in Figure 6.1.9, comprises a self-contained freshwater flow system (called the freshwater lens because of its general shape), floating on a sea-fed, saline ground-water base. There are two possible locations for the repository--in the lens of freshwater circulation and in the deep, near-static saline ground water - shown as A and B in the figure.

Geographically, three classes of island have been identified:

- Continental Islands - located on the continental shelves and including igneous, metamorphic, and sedimentary rock types

- Oceanic Islands - located in ocean basins and primarily of basaltic rock of volcanic origin

- Island Arcs - located at margins of oceanic "plates", primarily of tectonic origin, and frequently active with andesitic lavas.

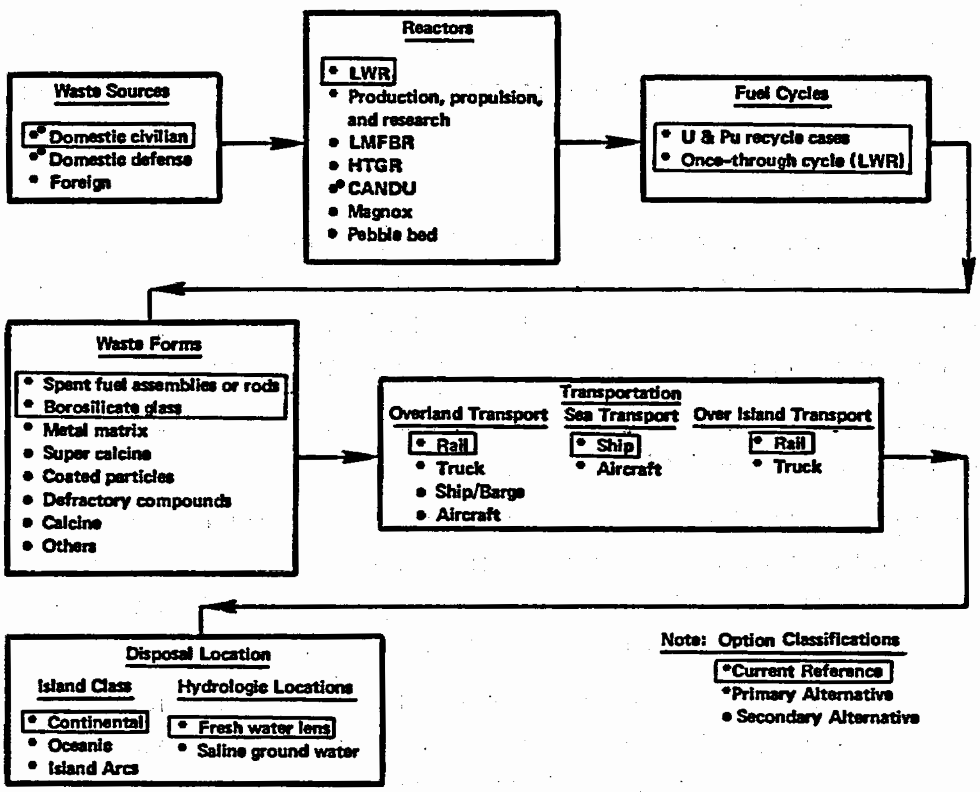


FIGURE 6.1.7. Major Options for Island Disposal of Nuclear Waste

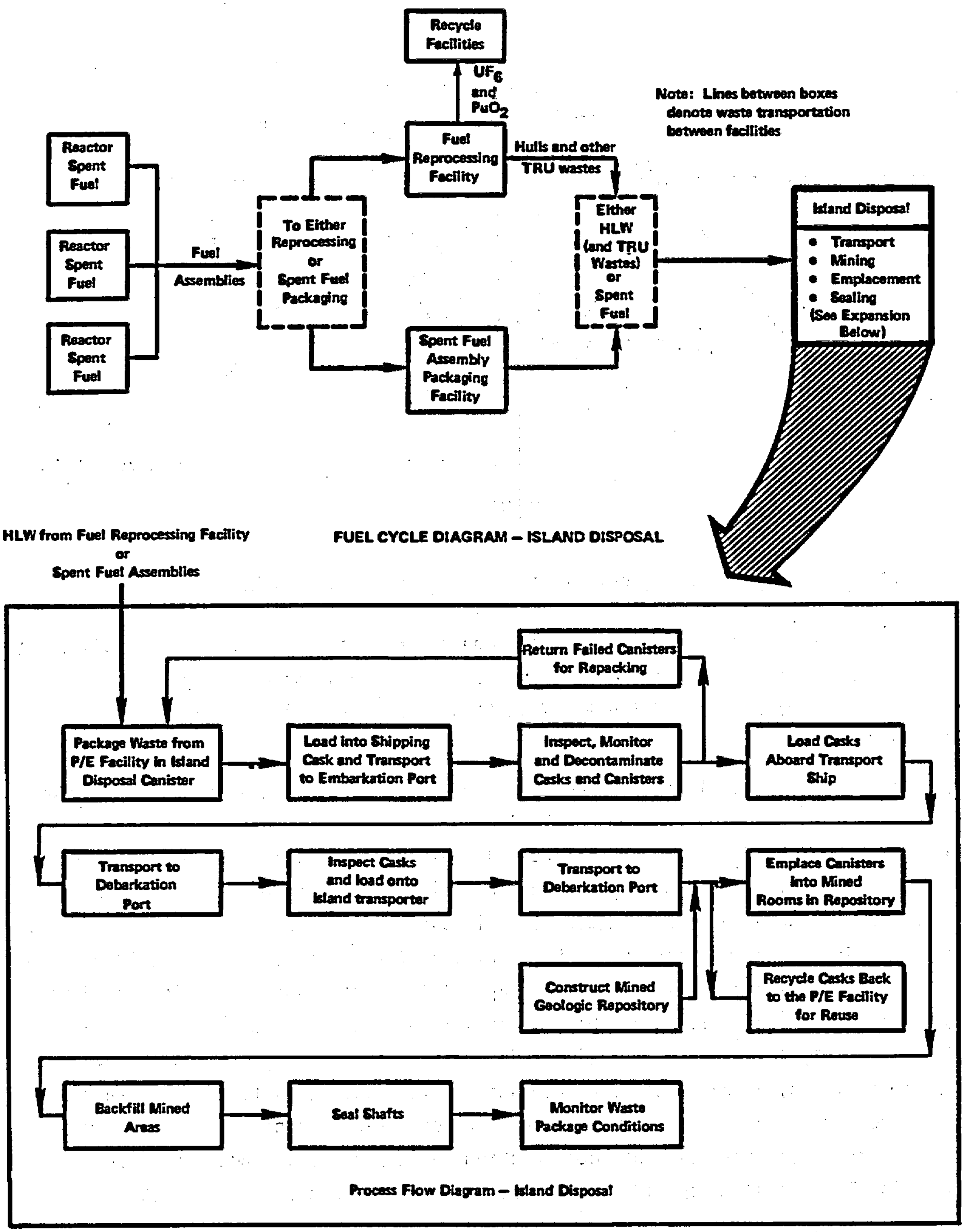


FIGURE 6.1.8. Waste Management System--Island Disposal

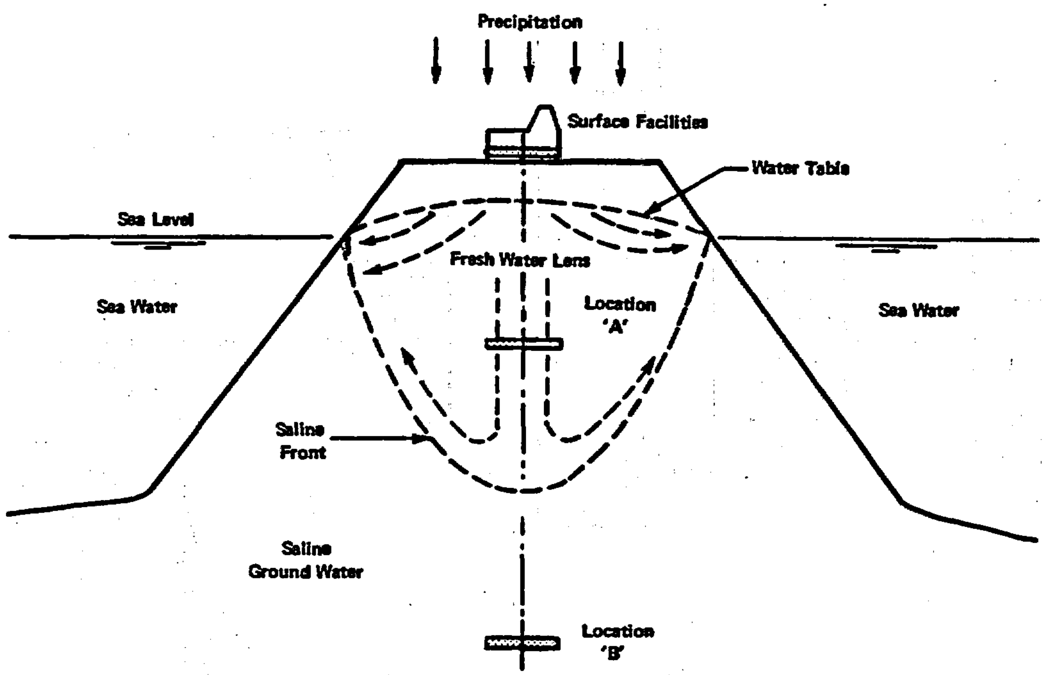


FIGURE 6.1.9. Hydrological Classification of Repository Locations

All three classes exhibit the classical island geohydrology described above, as modified by local geology and geographic setting. There are further discussions of the geology and hydrology of typical islands in DOE (1979), Todd (1959), Bott (1971), and Bayley and Muehlberger (1968).

Transportation Features. The island concept would incorporate the same basic procedure for transportation and handling as mined geological disposal. Of course, additional transportation from the mainland port to the island and additional receiving and handling facilities would be required. Transportation from the fuel reprocessing plant to the disposal site would be accomplished in three stages. The first stage would consist of truck or rail transport to a mainland port. Waste would be carried in transport casks that would cool the wastes and provide radiation shielding. (See Chaper 4 for a discussion of this procedure.) The second transport stage would be by ship to the island port. The subseabed disposal option (Section 6.1.4) details the operational features of this transportation phase. The casks would be cooled by either a closed-circulation water system, filtered forced-air system, or heat exchangers cooled by seawater. The coolant would be continuously monitored for radiation and temperature changes. Ship construction would provide for additional cooling. The ships could also include a shielded cell facility for examination of the casks.

The receiving port at the island would have the same features as the embarkation port described in Section 6.1.4. It could have a facility for temporary waste storage and transfer of the waste to specially designed transportation casks for final transport to the repository, the third phase. Conceptual design studies for island disposal are unavailable, but the required additional transportation facilities might be based on those discussed for the port and sea transport parts of the subseabed disposal option in Section 6.1.4.

Repository Facility. The layout of the reference repository for island disposal is a preliminary adaptation of the conventional geologic disposal concept discussed in Chapter 5. It is assumed that the island bedrock is crystalline and that the waste is emplaced approximately 500 m underground.

The conceptual design for an island crystalline rock repository is not supported by a data base comparable to that for salt repositories. The crystalline rock conceptual design discussed in Chapter 5 is assumed to be applicable to the underground aspects of island disposal except salt stockpile handling equipment would not be needed. The surface facilities for island disposal are assumed to be the same as for conventional mined geologic disposal.

Assuming that the repository capacity for spent fuel disposal is the same as for the conventional mined geologic disposal and that sufficient intermediate storage and transportation capacity can be provided, the once-through cycle would require four to eight island repositories, depending on the media. More respositories would be needed if island area were insufficient to support a repository of the size discussed in Chapter 5. Uranium-plutonium recycle wastes would require six to ten island repositories, depending on the island media (DOE 1979). The scheduled availability of the repositories for wastes from both fuel cycles would be expected to be a few years behind that of the conventional mined geologic disposal program.

Retrievability/Recoverability. Retrievability of emplaced waste or spent fuel from the rooms would be essentially the same as for the conventional mined geologic repository in crystalline rock. If retrieval were required because of deterioration or failure of the waste containers, special transportation containers and storage facilities would be needed. This need could be met by using a special cask design suitable for either rail, truck, or sea transport. Recoverability would also be similar to that with mined geologic disposal and would involve techniques similar to those used for the original emplacement process. Retrievability from island repositories could be complicated by the hydrogeologic characteristics of the sites.

Sealing, Decommissioning, and Monitoring. The sealing concepts might be the same as those for conventional mined geologic disposal in crystalline rock. The principal difference would be in the supply of labor and materials, which would involve sea transport to the island.

Final decommissioning of the island facilities could involve underground disposal of all contaminated equipment, the removal or disposal of all surface facilities, and suitable restoration and landscaping of the island.

Monitoring systems would be used during emplacement operations to detect air, surface water, and ground-water contamination. After the repository was sealed, a long-term monitoring system would be implemented. This system would be similar to those for the conventional geologic disposal concept, with modifications to suit the island option.

6.1.3.3 Status of Technical Development and R&D Needs

Present State of Development

In general, conventional mining techniques would be applicable to island repository construction. Transportation, storage, and handling requirements would be similar to those for the conventional mined geologic disposal concept, with the addition of the sea transportation link. Construction methods for ports would employ standard engineering practice.

Because the island disposal concept is so similar to the mined geologic repository option, the state of development is about the same. The ship loading and unloading requirements are similar to those described in the subseabed alternative, so again, the state of development is about the same.

Technical Issues

Technical issues that differ from those for mined geologic repositories lie in the areas of unique island hydrology and the resultant impacts of fresh or saline water on the package materials and the waste formulation.

For example: Is the waste form proposed for conventional mined geologic disposal appropriate for island disposal? Are the canisters that encapsulate HLW or the canisters of spent fuel compatible with the island repository environment? Should emplacement be in the freshwater zone or the saline ground-water zone?

Because a major incentive for considering island sites is a particular hydrological regime that frequently exists beneath them, efforts would be needed to:

- Verify the existence of a freshwater lens at various sites and determine its size.
- Determine the flow patterns and velocities of saline ground water at depths beneath the freshwater lens.
- Verify the stability of the freshwater lens in terms of the equilibrium between deep groundwater flows, salinity diffusion, precipitation and surface hydrology, the effects of sea level slopes, and other relevant processes in the natural state.
- Examine the perturbation to the lens caused by construction of the repository shafts and underground facilities, using simulation models and field evidence, if available. The shafts and facilities will tend to provide a sump that will drain either the freshwater or the saline ground water, depending on the location and depth of the repository.
- Examine the effects of heat generation on lens stability using simulation models. Heat may cause thermal convection cells that could flow counter to the freshwater circulation and modify the discharge pattern into the seawater.

R&D Requirements

To resolve these technical issues, specific R&D programs would be directed toward:

- Development of a system data base
- Study of hydrogeological aspects of island sites

- Development of criteria for and categorization of siting opportunities
- Risk assessment.

Implementation Time and R&D Costs

The time to complete the R&D, and the associated costs would be very similar to time and costs for a mined geologic repository. Increased R&D cost for the island concept would be expected to be a very small increment when compared to total costs for development of the mined geologic repository.

Summary

Major uncertainties, shortcomings, and advantages of the concept are summarized below:

- The transportation requirements to a remote location add to the overall risk of the concept.
- The state of knowledge relating to the hydrologic regime, upon which the concept relies, is not currently sufficient for siting or performance analysis.
- Considerable effort might be required to develop specialized waste forms and packages, if current reference concepts are not suitable.
- The approach does appear to be technically conservative if the hydrology is as predicted and to be capable of implementation in a step-wise manner.
- The concept employs the multi-barrier approach and has the additional attractive benefit of being remote.

6.1.3.4 Impacts of Construction and Operation (Preemplacement)

Impacts of construction and operation of predisposal systems in the island concept would be similar to those discussed in Section 5.6 for the mined geologic repository. Additional impacts from the sea transportation link and the port facilities would also be involved and are discussed in Section 6.1.4.4 for the subseabed disposal option. Impacts of mainland disposal are not discussed here.

Ideally, any island chosen for disposal would be totally uninhabited prior to construction of the repository (Selvaduray et al. 1979). In this case, the only non-occupational people impacted by construction and operation of the island repository would be families of those working at the facility.

Health Impacts

Radiological Impacts. Increased radiation exposure of occupational personnel under both normal and abnormal conditions would result from unloading of the waste at the receiving port, temporary storage of the waste, and transfer of the waste to the repository. Quantitative estimates of these exposures are not available at this time. However, unloading of the waste would probably result in exposures similar to those encountered during loading at the embarkation port, as discussed in Section 6.1.4.4 for the subseabed option. In addition, it is significant that the island repository would accept TRU wastes. This means that transportation impacts would be slightly greater than those for the subseabed option.

Moreover, although transportation-related impacts might be higher for island disposal, mainland benefits would be significant because of the elimination of the need to dispose of TRU wastes on the mainland.

The operation of the island repository itself is expected to be essentially the same as that for a mined geologic repository. Therefore, the exposure of occupational personnel to radiation should also be essentially the same. This exposure, during both normal and abnormal conditions, is discussed in Section 5.6.

In the event that there were any nonoccupational people on the island, the maximum dose received by any one of those individuals is expected to be similar to that received as a result of the operation of a mined geologic repository. However, because only a limited number of nonoccupational people should be present, total nonoccupational radiological health effects for an island repository are expected to be considerably less than those for a mined geologic repository.

Nonradiological Impacts. As indicated, impacts for island disposal should be similar to those of the subseabed and mined geologic disposal options. However, for an island repository in a relatively uninhabited area of the world, impacts would be significantly different from those of the mined geologic repository. In that case, potential non-occupational impacts would result primarily from transportation activities. Most transportation-related impacts are expected to be similar to those from the subseabed disposal option and are described in Section 6.1.4.4. That option, however, would not involve unloading waste material and increased transportation that could cause additional impacts from island disposal.

Natural System Impacts

Investigation of candidate island disposal sites would involve drilling and geophysical surveys, both on the island and in the adjoining offshore areas. During these activities, natural and wildlife habitats could be disturbed. Access and exploration operations could pollute both freshwater and seawater sources. Ecological effects could also arise from the use of explosives for seismic surveying. These impacts could be minimized by identification of sensitive areas and adequate planning.

Other ecological impacts, such as those described for the mined geologic repository in Section 4.8, would occur on the island selected for final disposal. However, because of the delicate balance of an island ecosystem, these impacts might require special consideration. In addition, the construction and operation of the required transportation and repository facilities would potentially impact the marine environment. These types of impacts have not been extensively evaluated.

Another important consideration is that small island ecosystems provide no refuge for the biota and ecosystems are much more easily affected by large-scale human activity. Furthermore, after the operational phase had ended, recolonization from outside sources would be far more difficult, and would take longer, than for a continental region. Finally, the types of species that recolonize an island could be expected to establish considerably different trophic structures than were present prior to construction.

Emplacement operations in the repository would be similar to those for the conventional mined geologic disposal concept. However, if an accident were to occur within the island repository, water might be present because of drainage into the excavation. Thus, these operations, and other activities associated with the island repository, could affect the freshwater regimes on the island. In addition, water pumped from the underground excavation would be brackish if the repository were located below the freshwater lens in the saline zone. Therefore, care would be required to prevent contamination of surface freshwater streams and lakes. Disturbance of the natural ground-water regime could result in some freshwater wells becoming saline. Such activity could significantly affect the island's ecosystem, of which freshwater is a critical element.

Socioeconomic Impacts

Construction of an island repository would require assembling and transporting a large work force to a remote island. These activities would affect the socioeconomic structure of coastal communities through which the project personnel and equipment were transported. Detailed assessment of these impacts has been limited, but information presented on the subseabed and ice sheet options provides a useful perspective (Sections 6.1.4.5 and 6.1.5.5).

On the island, socioeconomic impacts would be a different type of concern associated with the entirely new communities that would normally be established. Selecting unoccupied islands for a final repository would greatly reduce socioeconomic impacts.

Aesthetic Impacts

Aesthetic impacts of the island disposal option would be limited because few people would live in the vicinity of the repository. During construction and operation, authorized site personnel would be the only individuals to perceive aesthetic impacts.

Aesthetic impacts would also be associated with transportation activities. Although these are generally not viewed as significant, additional discussion on this matter appears in Sections 6.1.4.5 and 6.1.5.5 on the subseabed and ice sheet disposal options, respectively.

Resource Consumption

Construction and operation of the island repository facilities would require energy, as would transporting the waste material to the disposal site, over mainland, ocean, and island routes. There are no studies available to quantify these energy needs.

Although the size of the facility and the land area required would be similar to that for the conventional mined geologic concept, it should be recognized that island repositories would likely require that an entire island be devoted to a waste repository. This commitment of land might not be important, however, considering that extensive study would be completed before an individual island was proposed as a disposal site.

International and Domestic Legal and Institutional Considerations

The island disposal option, like the subseabed and ice sheet options, would require transporting waste material over the ocean, and the general international implications of such transportation are important. Emphasis in this discussion is placed on aspects unique to island disposal.

Two, possibly complementary, international considerations would have to be studied for island disposal. On the other hand, an initital motivation for island disposal is that it could provide an international repository for use by many countries. On the other hand, the siting of a repository on an island over which the U.S. does not have soveriegnty would require the approval of the nation that does.

International concerns could arise from countries in the vicnity of a proposed island repository. For example, if a remote island in the South Pacific were selected for an island repository, nations bordering the South Pacific might feel they were exposed to risks while receiving little or no benefit. Regardless of whether specific treaties were required, nations adjacent to any island disposal site ould be likely to voice concern and seek international assurance of the safe operation of these facilities.

6.1.3.5 Potential Impacts Over Long Term (Postemplacement)

Potential Events

As in land disposal of radioactive waste, island disposal would require careful assessment of the processes by which the radionuclides could migrate from the containers through the various barriers to man's environment. Actual island emplacement of any quantity of such waste could occur only after the completion of a program to demonstrate, by analysis and experiment, the retention capabilities of each of the natural and man-made barriers to migration.

Waste Encapsulation. The waste form and canisters used for island disposal might be similar to those used in a mined geologic repository on the mainland. Studies of the specific effects of ground-water chemistry in either the freshwater lens or deep saline zones would provide data for establishing leach rates in the crystalline rock site.

Ground-Water Transport, Freshwater Lens Location. Waste emplaced in the freshwater lens might be exposed to the very slow ground-water circulation within the lens. The velocities would depend on rock permeabilities, porosities, precipitation, and surface hydrology. A simplified conceptual view of the potential pathways and barriers is shown in Figure 6.1.10.

Waste in the freshwater lens circulating system might be expected to discharge at the shoreline. Natural ground-water flow patterns might be affected by thermal convection and repository construction. Concentrations at the exit zone have not been estimated.

Radionuclides might be sorbed by the host rock, which would substantially retard the waste transport within the lens. Sediments that might exist at the shoreline in the discharge zone could have useful sorption properties and retard radionuclides prior to discharge and dilution in the seawater.

Ground-water Transport, Saline Zone Location. It has been suggested that offshore islands may have essentially static saline ground water at depth, due to the absence of hydraulic gradients at sea level. However, the residual or continuing effects of oceanographic, geothermal, climatological, or other changes may create flow. These effects would need to be examined prior to siting a repository in such a location (see Figure 6.1.11).

Flow transport in the saline zone may be accompanied by dispersion and diffusion, which would result in reduced concentrations at a distance from the repository. The amount of sorption of radionuclides in the host rock or on seabed sediments would depend on the particular radionuclide, ground-water, and rock or sediment chemistry.

Seawater Contamination. It appears that the principal discharge of wastes from an island repository would be into the seawater, possibly through sediments. Discharge might occur in a relatively concentrated near-surface zone if the waste were located in the freshwater lens. This could cause contamination of littoral and near-surface aquatic systems.

Discharge from wastes located in the saline ground-water zone would likely be dispersed through the seabed if the thermal-convection effects were insufficient to distort the flow patterns significantly.

Volcanism. Some islands, particularly those in island arcs and to a lesser extent oceanic islands, are frequently highly active seismically and volcanically. Such activity could discharge the waste in either lava flows or into the atmosphere. Geologic data for the most recent volcanic event would be relied upon to establish inactivity before an island was selected as a disposal site.

Potential Impacts

In determining the potential impacts of island disposal over the long term, the following factors would be considered:

- Corrosion, leaching, and transportation of radionuclides to the biosphere by the ground water
- The influence of thermal effects on flow
- Thermal/mechanical effects on permeability and porosity
- Retardation of radionuclides on rock fractures and seabed sediments
- Sediment and current movements
- Pathways to man via marine organisms, typical marine activities, and island considerations.

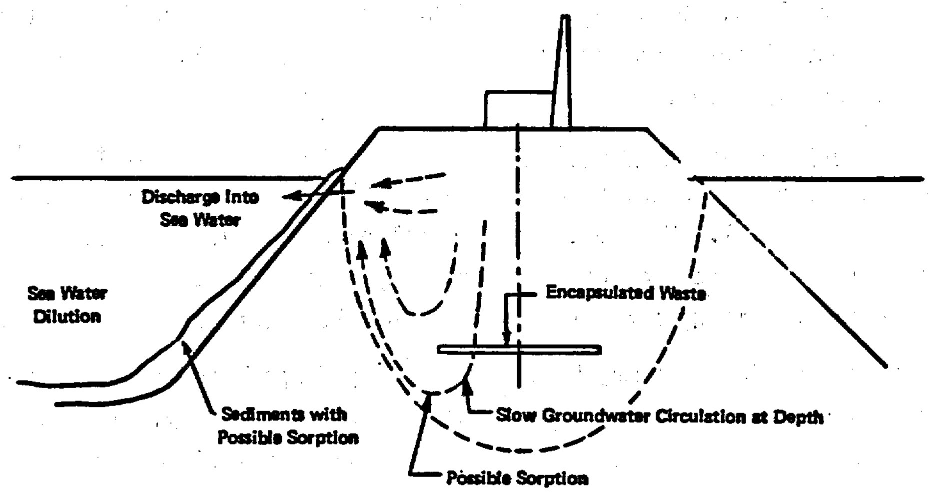


FIGURE 6.1.10. Isolation Barriers for Freshwater Lens Location

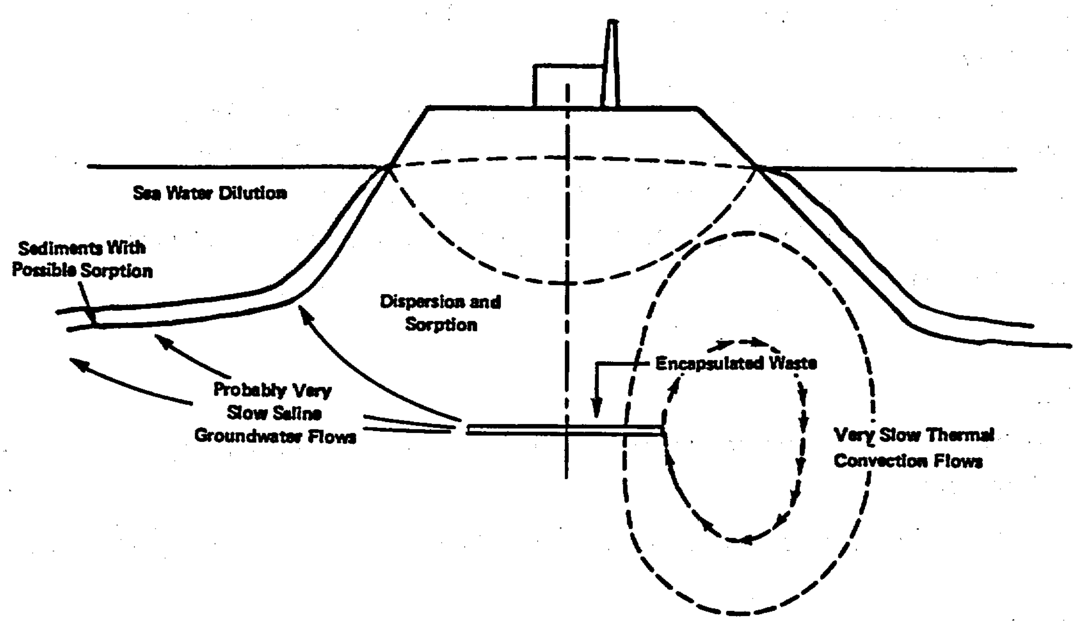


FIGURE 6.1.11. Isolation Barriers for Saline Zone Locations

Quantitative estimates of these impacts for the island disposal concept are unavailable at this time. However, it is expected that they would be similar to, but probably less significant than, those from a mined geologic repository. The reasons for the probable lessened impact are that (1) seabed sediments might provide significant sorption of certain radionuclides, (2) the sea would provide substantial dilution of discharges from the ground water, and (3) the island population, which would bear the greatest impacts, would be expected to be small in the long term because of the remoteness, size, and limited potential for inhabitation of any island that would be selected.

6.1.3.6 Cost Analysis

Detailed costs for island repository construction, operation, and decommissioning have not been estimated. It is estimated, however, that the cost of an island repository would be at least double that for a continental mined geologic repository because of sea transportation, the associated loading and unloading facilities, and the high salaries necessary for remote locations.

6.1.3.7 Safeguard Requirements

With the exception of ocean transportation, safeguard requirements for this concept would be expected to be similar to those for the mined geologic repository concept. However, the risk of diversion for the island disposal concept is primarily a short-term concern because of the remoteness of the disposal site and the major operational and equipment requirements for retrieval. Physical protection of the sensitive facilities and transportation operations would be the most effective way to deny access to the waste for the short term. For additional discussion of predisposal operations safeguards see Section 4.10.

6.1.4 Subseabed

6.1.4.1 Concept Summary

In subseabed disposal, wastes would be emplaced in sedimentary deposits of the ocean bottom that have been stable for millions of years. These deposits have a high sorptive capacity for the waste species (except for iodine and technetium) that might leach from the waste packages. Transport from ocean depths for any waste species escaping the sediments to the biologically active near-surface waters is expected to be a slow process that would result in dilution and dispersion. In addition, the great depth of the water column would constitute a barrier to human intrusion.

A program has been under way since 1973 to assess the technical and environmental feasibility of this concept for disposing of high-level nuclear wastes (Bishop 1974-75, Talbert 1975-78). The total seabed represents about 70 percent of the surface of the planet (of which less than 0.0001 percent would be used) and contains a wide variety of geologic formations. Theoretically, all wastes from the once-through cycle and uranium-plutonium recycle options could be emplaced in subseabed formations. But, because of volume considerations, other methods of disposal may be more practicable for contact handled and remotely handled TRU wastes.

The reference subseabed geologic disposal system for study purposes is the emplacement of appropriately treated waste or spent reactor fuel in a specially designed container into the red clay sediments away from the edges of a North Pacific tectonic plate, under the hub of a surface circular water mass called a gyre (mid-plate/gyre:MPG). (However, selection of the North Pacific as a study area in no way implies its selection as a candidate subseabed disposal site.) The reference method uses a penetrometer[a] for emplacing wastes in the sediments in a controlled manner that allows subsequent monitoring. A specially designed surface ship would transport waste from a port facility to the disposal site and emplace the waste containers in the sediment. A monitoring ship, which would completely survey the disposal site before operations began, could determine the locations of individual disposal containers and monitor their behavior for appropriate lengths of time. The ship would also maintain an ongoing survey of the surrounding enviromment.

(a) A penetrometer is a needle-shaped projectile that, when dropped from a height, penetrates a target material. It can carry a payload of nuclear waste and instruments designed to measure and transmit its final position and orientation relative to the sediment surface. Penetration depth is controlled by the shape and weight of the penetrometer, its momentum at contact with the sediment, and the mechanical properties of the sediment.

6.1.4.2 System and Facility Description

System Options

The reference concept for the initial subseabed disposal of nuclear waste has been developed from a number of options available at each step from the reactor to disposal in the subseabed repository.

Various options to be considered for the subseabed concept are summarized in Figure 6.1.12. The bases for selection of options for the reference concept are detailed in sources cited in Appendix M.

Waste-Type Compatibility

It is assumed for the reference case that subseabed disposal is limited to disposing of spent fuel, HLW and cladding hulls. Other wastes are assumed to be disposed of in a mined geologic repository. However, it should be noted that these wastes may also be appropriate for subseabed disposal if there are sufficient economic incentives.

Waste-System Description

The reference concept design was selected as a feasible approach based on available information and data and is not supported by a detailed system engineering or cost analysis. The waste-management system, including the fuel cycle and process flow, for the reference concept is shown in Figure 6.1.13.

Subseabed disposal has as its foundation a set of multiple barriers, both natural and man-made, that would be employed to ensure the safe isolation of nuclear waste. These barriers are (Bechtel 1979a):

- The waste form
- The waste canister
- The emplacement medium (i.e., sediment)
- The benthic boundary layer
- The water column.

The water column is a barrier primarily to intrusion by man, although it would provide dilution and dispersion for radioactive species.

The waste form (leach-resistant solid) and the metallic waste canister or overpack would be man-made barriers. It is assumed that they could be engineered as a multibarrier system to contain the waste for a period during which the heat-generation rate due to fission product decay would decrease to low levels.

The emplacement medium (clay sediment) shows evidence that it could provide long-term containment of the nuclides through its sorptive qualities, ion-exchange characteristics, and very low permeability.

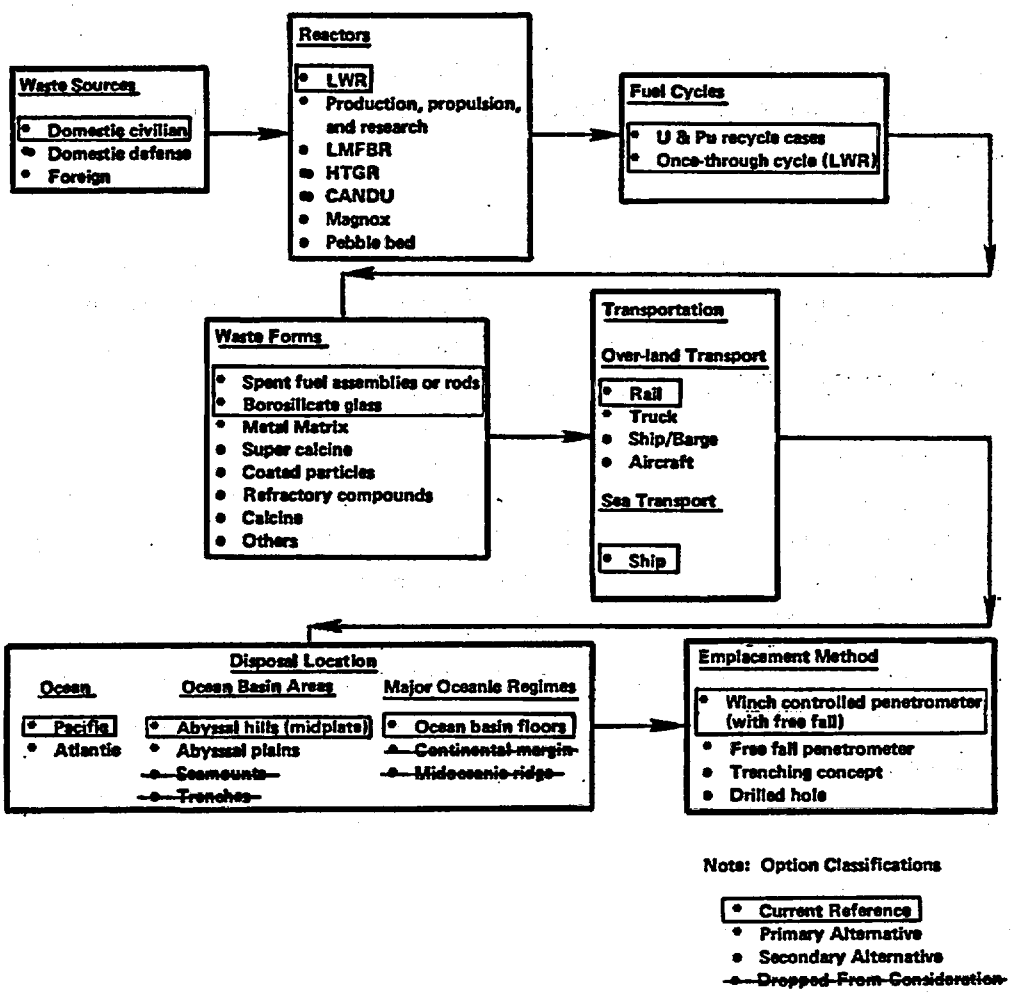


FIGURE 6.1.12. Major Options for the Subseabed Disposal of Nuclear Waste

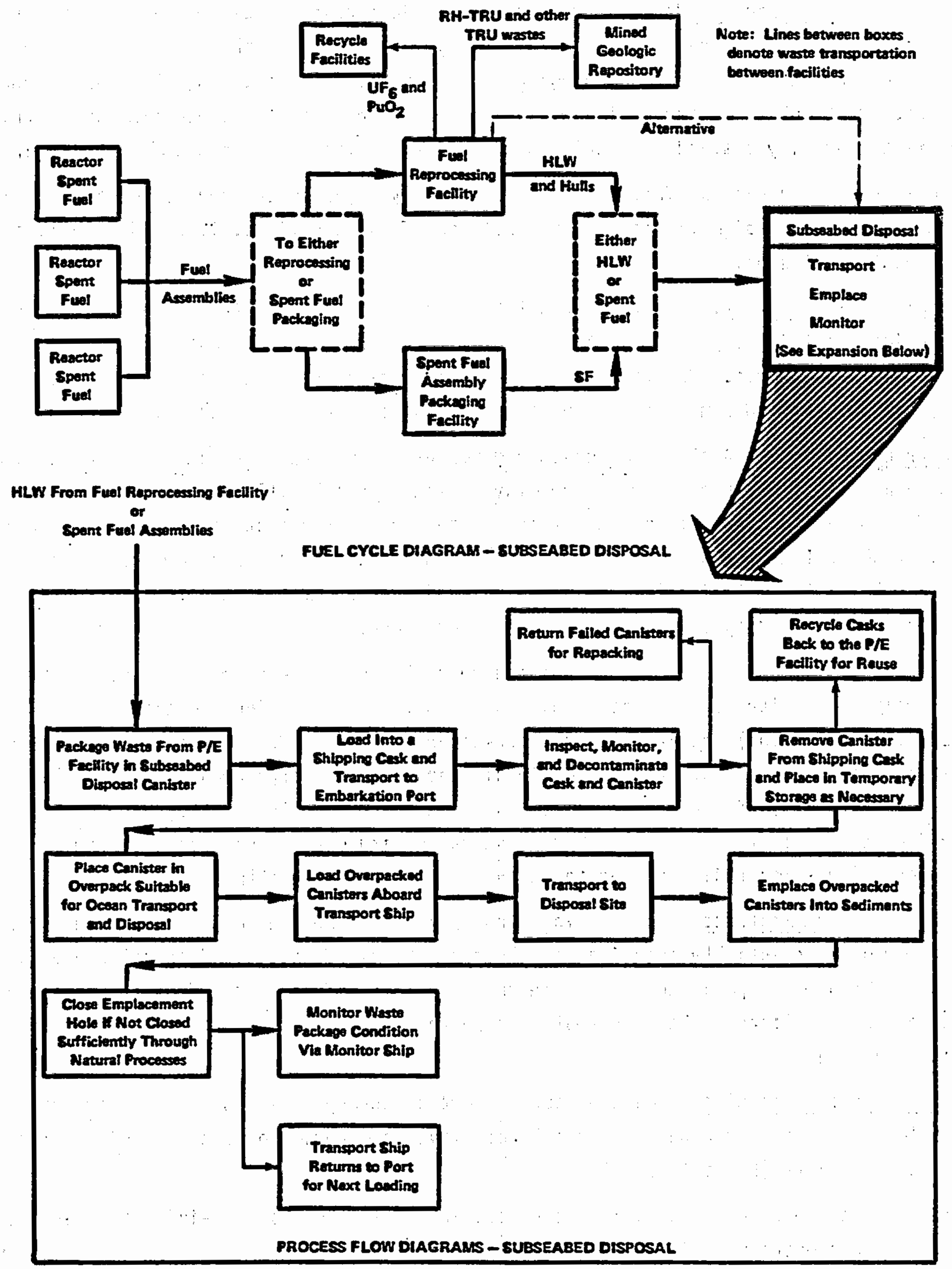


FIGURE 6.1.13. Waste Management System--Subseabed Disposal

The ocean's benthic boundary layer extends from less than 1 m below the sediment-water interface to 100 m above that interface. This layer results from the turbidity induced by natural flow processes and by the biological activity at, or just below, the sediment-water interface. Particulate matter, which would act to sorb radionuclides escaping the sediments, is temporarily suspended in this layer and then returns to the sediment surface.

The water column extends from the benthic boundary layer to the surface of the water. It would provide dilutional mitigation to the release of radionuclides. It would also be a barrier to man's intrusion.

Predisposal Treatment. The predisposal treatment of waste for the subseabed concept would be identical in many respects to the predisposal treatment of waste for the mined geologic repository concept. Chapter 4 of this document discusses the predisposal systems for both spent fuel and HLW common to all of the disposal concept alternatives.

Ocean Environment. Analysis of ocean regimes has shown that the most appropriate areas for subseabed waste containment would be clay-covered abyssal hill regions away from the edges of subocean tectonic plates underlying large ocean-surface currents known as gyres. These vast abyssal hill regions are remote from human activities, have few resources known to man, are relatively biologically unproductive, have weak and variable bottom currents, and are covered with red clay layers hundreds of meters deep.

These clay sediments are soft and pliable near the sediment-water interface and become increasingly rigid with depth. Tests have shown that they have high sorption coefficients (radionuclide retention) and low natural pore-water movement. Surface acoustic profiling indicates that such sediments are uniformly distributed over large areas (tens of thousands of square kilometers) of the ocean floor. As shown by core analysis, they have been continuously deposited and stable for millions of years, giving confidence that they would remain stable long enough for radionuclides to decay to innocuous levels (DOE 1979).

Transportation Features. The overland transportation features of the subseabed disposal concept would be essentially identical to those of the mined geologic disposal concept. In addition, subseabed disposal would require transportation of the waste from the mainland to the subseabed repository. The principal transportation requirements would be for seaport facilities and seagoing vessels.

a. Seaport Facilities. The subseabed reference concept assumes that seaport facilities would be used only for waste disposal activities and would not share services with other commercial endeavors (Bechtel 1979a).

The seaport would have facilities for receiving railway casks containing the waste canisters and for storing them in a water pool until shipment to the repository site. All required handling equipment, including that needed to load the canisters into seagoing vessels, would be available at the port.

The port facility could receive and handle 10,200 spent fuel canisters a year (Bechtel 1979b). For handling high-level reprocessing waste, the total annual throughput would be:

	Canisters
HLW	2,380
Cladding Hulls	2,300
End Fittings	1,520
Total	6,200

Cladding hulls and end fittings are not thermally hot. However, they would be handled in the same manner as HLW for storage and disposal because of their high radiation levels and the possibility of contamination by transuranic elements.

The shipping area of the port facilities would include a canister transfer pool and a transfer cask storage area. To load the ship, the canisters would be moved from the cask and transferred to the ship by crane. The dock facilities would accommodate two ships of the class described below.

b. Seagoing Vessels. Because of the quantities of waste canisters to be disposed of, subseabed disposal would require special dedicated ships (Bechtel 1979a). Each ship would contain equipment for handling the canisters during loading, a water pool to store the canisters during transportation, the necessary equipment to emplace the canisters in the sediment, and water cooling and treatment facilities.

The waste ships could have double hulls and bottoms. Waste canisters would be secured in the holds of the ships in basins filled with water. This concept of transporting fuel canisters in a shipboard storage pool, while new, is considered entirely feasible and is assumed for the reference study.

Disposal of spent fuel might require approximately 15 days to load a ship, 15 days for the round trip from port to repository, and up to 50 days to emplace the canisters at the subseabed site. Thus, a ship would make four trips a year. Based on transporting 1,275 canisters per trip, two ships would be required.

The sea-transportation requirements for HLW would be the same as those for spent fuel assemblies. It is estimated that the same numbers and class of ships as described above would be adequate for transporting HLW and cladding hulls. The same number of trips would be required, but total turnaround time would be about 15 days less because fewer canisters would be handled.

In addition to the ships used for the disposal operations, a survey ship would monitor the emplacement of canisters and their positions relative to one another.

Emplacement. It is assumed that a free-fall penetrometer would provide one alternative method for emplacing canisters in the seabed sediment (Bechtel 1979a). The canisters would have a nose cone to aid penetration and tail fins for guidance. Alternatively, they might be lowered to a predetermined depth and released, and would be designed to penetrate about 30 meters into the sediment. Laboratory tests indicate that the holes made as the canisters entered the sediment would close spontaneously. Canister instrumentation would permit a monitoring crew to track each canister to ensure proper penetration into the sediment and spacing between canisters.

The total seabed area required would be 560 km^2/yr (215 mi^2/yr) for HLW and 920 km^2/yr (354 mi^2/yr) for spent fuel assemblies, based on an arbitrary spacing of 300 m (984 ft) between canisters and a waste disposal system of 5,000 MTHM/yr.

Retrievability/Recoverability. Retrievability has not been designed into the system concept (though during the experimental period all emplaced radioactive material would be designed for retrievability) (DOE 1979c). Postemplacement waste-canister recovery from any of the four emplacement options (see Figure 6.1.12) would be possible with existing ocean engineering technology, but estimated costs are high.

Monitoring. After the wastes were emplaced, a monitoring ship would use instrumentation on the ship, on the ocean bottom, and on the canisters to determine information about the buried canister: e.g., its attitude and its temperature.
This monitoring would continue for as long as necessary to verify the performance of the subseabed isolation system.

6.1.4.3 Status of Technical Development and R&D Needs

Present State of Development

The status of concept design, equipment, and facilities for different facets of a subseabed disposal operation is described below.

Emplacement Medium. Properties of the red clay sediment of the ocean's abyssal hills have been studied extensively under the Subseabed Disposal Program (SDP) (Talbert 1977, Sandia 1977, Sandia 1980). The considerable data collected indicate that the sediment is a very promising emplacement medium. The SDP has collected data on nuclide sorption and migration, effects of heat and temperature, ecosystems, and other aspects of the subseabed environment in these sediment areas. The program was started in 1973, and studies of the emplacement medium and of concept feasibility are planned to be completed in 1986. After that, the program would deal with other engineering problems, such as the handling of waste during sea transportation and emplacement (Sandia 1980).

Emplacement Methods. The SDP has not yet defined the methods of waste emplacement in the subseabed. The technical problems associated with this task would be addressed after the studies on sediment properties are completed. In other words, the required depth of emplacement, spacing of canisters, method for assuming hole closure, etc., would have to be known before emplacement methods could be developed.

Four possible methods of emplacement are being considered: (1) free-fall penetrometer, (2) winch-controlled penetrometer descent to a determined depth and final propulsion (the reference concept), (3) trenching, and (4) drilling. The operations are described in Reference 4. The first two methods that use penetrometers present fewer technical challenges since the penetrometer is a widely used tool in marine, land, space, and arctic operations.

Waste Form. The waste form and the canister design required for subseabed disposal of spent fuel have not been determined. Because of the high hydrostatic pressures at the ocean bottom, one important characteristic of the waste package would be a filler material with low compressibility. Generally, metallic fillers would satisfy this requirement, but other solid materials could be more acceptable because of cost advantages, resource conservation, and easier process technology.

The waste form required for storage of HLW in a subseabed repository has not been determined. It is believed that borosilicate glass might be adequate, especially if the temperature of the canister-sediment interface were maintained below 200 C (392 F). This would require adjusting the age of the waste and/or the diameter of the canister to provide rapid heat flow away from the canister. Other waste forms are also being considered.

Waste Containment. Due to the expected effects of high heat and radiation on the properties of the subseabed sediments, waste containment would have to be maintained for a few hundred years to delay the release of nuclides. Experimental data on the rate of corrosion of metallic materials in hot brine and seawater, collected primarily to improve the material performance in desalinization plants and in geothermal applications, would add to the confidence that this capability can be provided.

The SDP has also included laboratory experiments with metallic materials subjected to a seawater environment of 200 C (392 F) and 1,000 psi (6.9×10^6 Pa). Plates of Ticode 12 showed the lowest rate of corrosion, as determined by a weight-loss technique (Talbert 1979).

Facilities. The seaport storage facilities and the facilities that would have to be built aboard ship have not been developed. However, the technology for building them is available since they would resemble existing facilities, such as spent fuel storage pools and ordinary port facilities. The seaport location, size, and capabilities are not yet defined by the SDP.

Technical Issues

The engineering aspects for subseabed disposal have not been established. The transportation logistics, regulations, and the appropriate transportation "package" have not been developed. The precise size and type of facilities that would be built are not known, and the time and motion studies to select the optimum ship size have not been made. In addition, a large area of uncertainty revolves around the methodology that would be used to emplace the waste. Techniques to ensure that waste canisters were placed deep enough into the sediment have not been demonstrated.

If demonstrated, a major attribute of subseabed disposal would be the ability of the sediments to hold radionuclides until they had decayed to innocuous levels. To determine whether these sediments could actually do this, the following technical issues would need resolution.

Ion Transport in the Sediment. More data would be required regarding the rates at which the radioactive ions transfer through the sediment. Studies and empirical data would be required to determine the thermal interaction with canister materials and wastes, conduction, and convection through the sediment.

Ion Transport to the Biosphere. The paths and rates at which the radioactive ions could transfer from the sediment, through the benthic boundary layer, and into the water column are not known. Both mathematical models and empirical experiments would be required to obtain this information. Modeling would also be required to determine a realistic rate of migration up the water column.

Sediment Mechanical Requirements. The subseabed sediments that would be candidates for nuclear waste disposal are between 4,000 and 6,000 m (13,000 and 20,000 ft) below the ocean surface. Further information would have to be acquired regarding their macroscopic (as well as microscopic) structural characteristics. These characteristics include sediment closure after emplacement and long-term sediment deformation and buoyancy resulting from heating.

R&D Requirements

The SDP is divided into seven R&D fields of study (see Sandia 1980), each with numerous subdivisions. As far as funding and the state of technology allow, all of these studies are being pursued simultaneously, though not all at the same level of detail. An eighth field, safeguards and security, would be established later as the results of the other seven studies become known. Brief descriptions of these eight studies which define R&D requirements, follow:

Site Studies. Current studies include evaluation of North Atlantic and North Pacific oceanic areas that meet site suitability criteria. From these areas, certain study locations have been, and will continue to be, identified for more intensified study.

Environmental Studies. Environmental studies include physical and biological oceanography. They focus on analyzing physical characteristics of the water column from the ocean surface to the sediment surface, and on gathering all pertinent information about the marine life that inhabits the water column. The ultimate purpose of these studies is to determine whether, and to what degree, the physical and biological characteristics of the ocean would accelerate or slow the transport of accidentally released radionuclides to man's environment.

Multibarrier Quantification. The multibarrier study includes the sediment, the canister, and the waste form, both immediately adjacent to the waste container and further afield, to determine their natural characteristics. Again, the ultimate purpose is to learn whether, and to what degree, they would allow released radionuclides to be transported. A second purpose is to learn how they would react to the heat and radiation generated by a waste container, as well as to any engineered modification to the sediment such as artificial closing of the emplacement hole.

Transportation. Transportation studies include four subdivisions:

- Land transport with investigations directed to transporting HLW and/or spent fuel from an originating plant to the port facility by rail, road, or barge.
- The port facility, including a receiving structure.
- The staging area, to include cooling facilities for holding waste packages until they could be loaded.
- Sea transport with studies including design of special transport/emplacement vessels and of travel routes designed to minimize interaction with shipping lanes and all other forms of maritime activity. It is likely that this would be a self-powered ship, but it could be a vessel that could be towed, possibly under water. Transportation technology is in early planning stages, pending determination of disposal feasibility.

Emplacement and Monitoring. The study of emplacement and monitoring focuses on the time period that begins when waste packages would be removed from their cooling area on the transport vessel and continues through burial deep in the subocean sediments and closure of the entrance hole, either naturally or artificially. An intrinsic part of the process would be the monitoring function. Monitoring would include surveying precise disposal locations, guiding emplacement mechanisms into those locations, and tracking the integrity, attitude, and stability of waste containers for as long as would be required after emplacement.

Social/Political Studies. Even if technological and environmental feasibility for the subseabed disposal concept were established, domestic and international institutions would ultimately determine whether the concept could be used. There are no laws or agreements at this time that specifically prohibit or allow subseabed disposal. Issues important to this area are further discussed in Section 6.1.4.4 under International and Domestic Legal and Institutional Considerations. International agreements and structures would enhance the implementation of the concept. Evaluation of the current political and legal postures of all countries that might be involved in subseabed disposal is under way. The existence of an international NEA/OECD Seabed Working Group is indicative of the international interest in the concept.

Risk/Safety Analyses. As data become available, risk and safety analyses would be completed on all aspects of the SDP.

Security and Safeguards. Except in the most general terms, studies in these areas would have to await data acquisition and assessment.

R&D Costs/Implementation Time

Research and development is assumed to end when the technology had been translated into routine practice at the first facility. Follow-on R&D in support of facility operation is considered in a different category.

To date, almost all resource expenditures have been focused on the technical and environmental feasibility of the subseabed geologic concept, rather than on specific on-site studies or demonstrations of current engineering practice. The estimated total R&D costs are $250 million (DOE, 1979).

The SDP program plan has been divided into four distinct phases (Sandia, 1980). In each phase, the concept feasibility is assessed. The estimated completion dates shown do not consider programmatic perturbances resulting from regulatory or institutional influences.

- Phase 1 Estimation of technical and environmental feasibility on the basis of historical data. Completed in 1976.
- Phase 2 Determination of technical and environmental feasibility from newly acquired oceanographic and effects data. Estimated completion date: 1986.
- Phase 3 Determination of engineering feasibility and legal and political acceptability. Estimated completion date: 1993-95.
- Phase 4 Demonstration of disposal facilities. Estimated completion date: 2000 to 2010 (Anderson et al. 1980).

Summary

Major uncertainties, shortcomings, and advantages of the concept are summarized below:

- The remoteness of the location, apparent sorption capacity of the sediments, and demonstrated stability of the site are attractive attributes.
- The concept could be implemented in a step-wise fashion.
- The expected performance of packages and waste form in the environment at the seabed is not well understood.
- Specific new domestic legislation and international agreement would likely be required.
- Retrievability to allow for corrective action purposes might be difficult.
- Transportation requirements to a remote location add to the overall risk of the concept.

6.1.4.4 Impacts of Construction and Operation (Preemplacement)

Health Impacts

Both radiological and nonradiological health impacts are discussed below.

Radiological Impacts. Both occupational and nonoccupational doses prior to the waste arriving at the seaport facility are expected to be similar to those anticipated for a mined geologic repository, as presented in Chapters 4 and 5.

The occupational and nonoccupational radiological impacts of the operation of the seaport facility and the seagoing vessels have been developed by Bechtel (1979a), and are presented in Table 6.1.11. These impacts are conservatively estimated as equivalent to those for away-from-reactor storage pools (AFR), corrected in consideration that:

- The primary waste handled at the subseabed facilities would be 10 years old.
- The primary waste at the subseabed facilities would be encapsulated.
- The number of personnel is expected to be smaller at the seaport facility than at the AFR facility. This may be offset by the fact that personnel might receive occupational doses for longer time periods while serving aboard ship.

TABLE 6.1.11. Radiological Impacts Of The Normal Operation At A Subseabed Repository

	Whole Body Dose, man-rem/yr	
	Spent Fuel	High-Level Waste
Occupational		
Seaport Facility	340	200
Seagoing Vessels	340	200
Nonoccupational		
Seaport Facility	40	10
Seagoing Vessels	Negligible	Negligible

Bechtel (1979a) gives the consequences of abnormal events at subseabed facilities. These consequences are equated with accidents postulated for the AFR (i.e., design basis tornado) facility for the most exposed public individual. No probability analysis was included. For spent fuel disposal, the radiological impacts of an abnormal event would be 0.02 mrem/event for the seaport facility and 0.003 mrem/event for the seagoing vessels. For HLW, these impacts would be 0.001 mrem/event and 0.002 mrem/event, respectively.

The maximum risk would be posed by the sinking of the seagoing vessel or by loss of waste canisters overboard. Except for accidents in coastal waters where mitigation actions could be taken, the radioactive materials released into the sea following such an event would disperse into a large volume of the ocean. Some radionuclides might be reconcentrated through the food chain to fish and invertebrates, which could be eaten by man. Bechtel (1979a) assumes that the waste could be retrieved if either event were to occur and does not provide an impact estimate. The doses provided in Table 6.1.12 for such an event are taken from EPA (1979).

Nonradiological Impacts. The numbers of injuries, illnesses, and deaths related to the construction and operation of the subseabed disposal option prior to the waste arriving at the seaport facility/repository are expected to be similar to those for the mined geologic options. At the seaport facility, it is estimated that the impacts would be no greater than those associated with surface storage and transfer facilities to be used with a reprocessing plant or spent fuel overpacking facility. These impacts are discussed in Chapter 4.

Additional areas specific to subseabed disposal that would have nonradiological health impacts are the construction of seagoing vessels and the conduct of operations at a seaport and on the ocean. Although there are no quantitative estimates of these impacts, it is anticipated that they would be similar to those incurred during the construction and operation of conventional seagoing vessels and operation of conventional dock facilities.

Natural System Impacts

Impacts to the natural environment for this disposal option would be related primarily to transportation and emplacement activities. Radiological concerns would be most significant

TABLE 6.1.12. Estimated Dose Commitment From Marine Food Chain For Loss of Waste At Sea

	Population man-rem	Average Individual, rem
Undamaged Spent Fuel		
Continental Shelf	510	5.9×10^{-4}
Deep Ocean	100	1.1×10^{-4}
Damaged Spent Fuel		
Continental Shelf	1×10^{5}	0.11
Deep Ocean	100	1.1×10^{-4}
HLW (Plutonium Package)		
Continental Shelf	Not provided	Not provided
Deep Ocean	100	1.1×10^{-4}

under abnormal conditions, while nonradiological impacts could also pose problems under normal operating conditions.

Transportation-related impacts for those activities occurring before the waste material was loaded on the ships would be similar to those for a mined geologic repository. Once the material was loaded onto the ships, impacts to the marine environment would have to be considered. In the case of potential accident conditions at sea, the design of the waste transporting vessels to include double hulls and bottoms would reduce the likelihood of releasing harmful material into the environment.

There are several uncertainties that limit the ability to predict natural system impact levels with confidence. Of primary concern is a lack of understanding of ion transport within the sediment and biosphere, including the benthic region, the water column and ocean life forms. In addition, the extent of the isolation barrier that the resealed sediment would provide after emplacement is not clear. Each of these factors makes detailed impact assessment difficult.

Other subseabed disposal impacts identified, but not quantified by Bechtel (1979a), include minor air emissions from construction equipment, dust generation, and road, rail, and vessel emissions. Construction-related impacts on water quality and vegetation as well as impacts on the marine environment resulting from dredging and breakwater construction could be locally significant. Although these impacts were identified by Bechtel (1979a), there are no data that indicate they would be significant.

Socioeconomic Impacts

Because a major land repository would not be required under this option, the most important socioeconomic impacts would be attributable to transportation activities. Transportation activities fall into three categories: (1) transportation of wastes on land to the port where the wastes would be transferred to the ship, (2) waste-handling activities at the port facility, and (3) ocean transportation from the port facility to the point where the material would be deposited in the seabed sediment.

Socioeconomic impacts would be concentrated at the point where support activities were most intense: at the port facility. The nature of the activity has led certain reviewers to conclude that one of the most significant factors associated with this disposal option would be difficulty in finding a suitable dedicated (Bechtel 1979a). Moreover, they project moderate community impacts and suggest that local socioeconomic impacts could reach significant levels.

Detailed projections of the impact of implementing this disposal option on the public and private sectors could be made only on site-specific basis. Nevertheless, impacts would be expected in the coastal area near the port facility. The total anticipated increase in employment for a 5000 MTHM per year disposal system, although quite concentrated, is expected to be less than 2000 people.

Aesthetic Impacts

The significance of aesthetic impacts would depend on the appearance and operating parameters of a facility, as well as on the extent to which it would be perceived by humans. For the subseabed disposal option, much of the waste-handling and trasportation activities would occur in remote areas of the ocean. Consequently, the aesthetic impacts, regardless of their nature, would not be significant.

Aesthetic impacts near the port facility, however, could be locally significant. Such impacts could be accurately determined only on a site-specific basis. However, it is important to recognize that the required port facilities for a nuclear waste handling facility would be substantial.

Resource Consumption

Use of energy and construction of seaport facilities and seagoing vessels would be the primary resource consuming activities in this option. Energy would be consumed during land transportation, loading, and sea transportation activities. A quantitative estimate of energy consumption is provided in Table 6.1.13.

The seaports would have facilities for receiving railway casks containing the waste canisters and for placing them in interim storage. Interim storage pools should be able to handle one-half of the anticipated yearly volume of wastes (2500 MTHM) and are expected to

TABLE 6.1.13. Estimated Energy Consumption

	Spent Fuel	HLW
Propane m^3	2.4×10^4	1.0×10^7
Diesel, m^3	5.0×10^6	1.6×10^6
Electricity, KWh	2.0×10^{10}	5.7×10^{10}

require an area within the boundaries of the port area subseabed support facilities of 2320 m^2 (25,000 ft^2) (Bechtel 1979a). Other storage and transfer facilities would also be needed. The total area required for all the required facilities is expected to be over 3600 ha (8500 acres).

Construction of the waste disposal ships with double hulls and bottoms, waste handling equipment for loading, and carefully constructed compartments for holding the wastes duirng transportaton activities, like construction of the port facilities, would lead to the consumption of steel and other basic construction materials. An estimate of the material consumption is provided in Table 6.1.14.

International and Domestic Legal and Institutional Considerations

The subseabed disposal option, like the island and ice sheet options, would require transporting waste material over the ocean, and the general international implications of such transportation are important.

Any implementation of subseabed disposal is far enough in the future that many current legal and political trends could change. However, it is not too early to identify important problems, so that possible developments could be foreseen and controlled.

The use of subseabed disposal would be governed by a complex network of legal jurisdictions and activities on both national and international levels. Domestic use of subseabed disposal of radioactive waste would require amendment of the U.S. Marine Protection, Research, and Sanctuaries Act of 1972 (The Ocean Dumping Act) which currently precludes issuance of a permit for ocean dumping of high-level radioactive waste.

Table 6.1.14. Estimated Material Consumption for Ship and Facility Construction (in MT)

	Spent Fuel	HLW
Carbon Steel	877,000	282,000
Stainless Steel	83,500	22,500
Components		
Chronium	14,200	4,600
Nickel	7,500	2,000
Tungsten	---	---
Copper	1,400	1,900
Lead	12,900	2,900
Zinc	1,200	600
Aluminum	13,000	1,400

The London Convention of 1972, a multinational treaty on ocean disposal, addresses the problem of dumping of low-level and TRU wastes at sea and bans the sea dumping of high-level wastes (Deese 1976). This treaty is currently being revised to deal more specifically and completely with the problem of dumping low-level and some TRU wastes. This treaty arguably does not preclude the controlled emplacement of high-level wastes or spent fuel into geologic formations beneath the ocean floor. However, the intended prohibition of the treaty would require clarification.

Subseabed disposal might offer the important political advantage of not directly impacting any nation, state, or locality. Likewise, the alternative might have the disadvantage of incurring risk to nations that do not realize the benefits of nuclear power generation.

Assuming that the real impact uncertainties associated with the subseabed concept were resolved, the primary political disadvantage of subseabed disposal would be its possible perception as an ecological threat to the oceans. If publics, governments, and international agencies were to view such disposal as merely an extension of past ocean dumping practices, implementation would be difficult if not impossible. However, if this option were understood as involving disposal in submarine geolgoic formations that have protective capacities comparable to or greater than similar formations on land, opposition might be less.

6.1.4.5 Potential Impacts Over the Long Term (Postemplacement)

Potential Events

Earthquakes, volcanic action, major climatological and circulational changes, and meteorite impacts are examples of natural processes that might affect subseabed containment stability. Careful selection of the ocean area would minimize the probability of the first three events occurring. There is no known method of minimizing the probability of meteorite impact other than concentrating emplacement, which, while reducing the random target area, would correspondingly increase the potential consequences if a meteorite did strike. On the other hand, other damage caused by any meteorite that could penetrate 5 km (3 mi) of water would make the release of emplaced radioactive waste insignificant.

For HLW disposed of in a subseabed repository, a very low probability for criticality is assumed because of the great distances between canisters at the bottom of the sea. For spent fuel, the probability of criticality might be somewhat greater because of the higher fissle content of a single canister.

Since the site would be located in a part of the ocean with no known materials of value, future human penetration would be highly unlikely.

Potential Impacts

Two models have been developed by Grimwood and Webb (1976) to characterize the physical transport and mixing processes in the ocean, as well as incorporation in marine food chains and ultimate consumption of seafood and radiation exposures to man. Although there is some question as to the applicability of these models to the subseabed disposal option, the following summary of results using these models is presented until such time as better estimates of radiation exposures to man from subseabed disposal are available.

The individual doses resulting from the consumption of surface fish, deep-ocean fish, or plankton are expected to be well below the maximum permissible levels. External individual doses[(a)] from contamination of coastal sediments are expected to be fractions of the ICRP dose limit for both skin and whole body irradiation. The largest annual internal population doses to the whole body and bone due to the consumption of surface fish would be about 4×10^4 and 10^5 man-rems, respectively. The largest annual external population doses from contaminated sediments would be about 10^3 to 10^4 man-rems for both skin and whole body. These large population doses would occur during the early stages of postemplacement and would decrease during the later stages.

As an attempt to provide a further yardstick against which to compare the results of the calculations, Table 6.1.15 gives the concentrations of nuclides predicted by the modeling, as well as the natural activity in seawater.

6.1.4.6 Cost Analysis

An estimate of capital, operating, and decommissiong costs for subseabed disposal has been made for both spent fuel disposal and HLW disposal (Bechtel 1979a). Both are based on penetrometer emplacement. All estimated costs are in January 1978 dollars.

TABLE 6.1.15. Levels Of Natural And Wastes Radionuclides In Seawater

Nuclide	Natural Activity In Seawater, Ci/cm^3	Max Widespread Surface Water Conc. Predicted From Postulated Waste Disposal Operation, Ci/cm^3 (No Containment)
Actinides		
Pb-210	$(1 - 9) \times 10^{-11}$	2×10^{-15}
Pb-210	1×10^{-10}	2×10^{-15}
Ra-226	1×10^{-10}	2×10^{-15}
Th-230	$(0.6 - 14) \times 10^{-13}$	2×10^{-17}
Th-234	1×10^{-9}	1×10^{-15}
U-234	1×10^{-9}	1×10^{-15}
U-238	1×10^{-9}	4×10^{-15}
Pu-239		1×10^{-12}
Fission Products		
H-3	2×10^{-10}	1×10^{-12}
Sr-90		4×10^{-10}
I-129	3×10^{-11}	3×10^{-14}
Cs-137		6×10^{-10}

(a) Based on world population

In each case, only those costs associated with and peculiar to subseabed disposal are addressed. Facilities common to all disposal options under consideration, such as transportation and geologic repository facilities, are not specifically addressed.

Capital Costs

The capital costs for the subseabed disposal alternative are categorized as follows.

Seaport Interim Storage Facility. This installation would provide receiving facilities for 5,000 MTHM/yr of spent fuel assemblies in 10,200 canisters. It would also be designed to provide interim storage for 5,000 canisters (2,500 MTHM). The same facility would receive the HLW and hulls from a 5,000 MTHM/yr fuel recycling system. Interim storage would be provided for 3,100 of these canisters at the port facility.

The seaport interim storage facility would be similar to a packaged fuel receiving and interim storage facility (Bechtel 1977) appropriately adjusted for size and waste form. The capital cost estimates are $240 million for the spent fuel case and $190 million for the HLW case.

Port Facility. The port facilities for both disposal cases are assumed to be identical for cost estimating purposes. The capital cost estimate is based on a recent estimate of another facility (Bechtel 1979a). The estimate for this port is $24 million.

Disposal Ships. The two disposal ships for the spent fuel case would have a capacity of 1,275 canisters each, while those for the HLW case would have a capacity of 775 canisters each. Since the canister capacity difference would be offset by the heat load and cooling requirement difference, the ships are assumed to be identical for estimating purposes.

The capital cost estimate of the ships is based on an estimate for a mining ship (Global Marine Development, Inc. 1979) appropriately adjusted. The estimated capital cost of the two disposal ships is $310 million ($155 million each). Note however that sophisticated offshore oil well drilling ships have been reported to cost between $50 million and $70 million each (Compass Publications 1980) or about half the above estimate.

Monitoring Ship. The capital cost for the monitoring ship was estimated from available data for oceanographic vessels. The estimate is $3.0 million for the ship and an additional $0.9 million for navigation and control, special electronics, and other surveillance equipment and for owner's costs. This brings the total capital cost to $3.9 million (Treadwell and Keller 1978).

Operating Costs

Operating costs for the subseabed disposal concept are estimated on a per year basis based on 5,000 MTHM/yr of both waste forms (spent fuel and HLW). This would result in virtually the same sea transportation requirements (number of trips per year). However, differences would occur for the HLW disposal case in years 1 through 9, when only hulls would be processed and disposed of, and during years 41 through 49, when only HLW would be disposed of.

The estimated yearly operating costs for the subseabed disposal concepts are presented in Table 6.1.16.

Operating costs associated with the reference subseabed disposal concept but also common to other disposal concepts are assumed to be similar. These costs would include transportation, AFR facilities (for the spent fuel), P/E facilities, and geologic repository facilities (assumed for the reference concept).

Decommissioning Costs

Decommissioning costs particularly associated with subseabed waste disposal operations would probably be limited to the seaport, interim storage facility, the port facility, and the disposal ships. The monitoring ship is not expected to be affected by radioactive waste during its 40 years of operation. Any decommissioning costs associated with the monitoring ship are assumed to be offset by its salvage value, which results in a zero net decommissioning cost.

The decommissioning cost of an AFR facility is used as the basis for the decommissioning cost of the seaport interim storage facility (Bechtel 1979b). These costs, based on 10 percent of capital cost excluding owner's cost, are approximately $23 million for the spent fuel disposal and approximately $18 million for the HLW disposal case.

The decommissioning costs for the port facility and two disposal ships are the same for both waste forms and are estimated to be about $2 million and $29 million, respectively, assuming 10 percent of capital cost less owner's costs.

Costs for decommissioning other facilities associated with subseabed disposal and common to other waste disposal alternatives are assumed to be similar. These facilities include AFR facilities (for the spent fuel), P/E facilities, and geologic repository facilities. These

TABLE 6.1.16. Estimated Operating Costs

Facility	Estimated Cost, $ million/yr Spent Fuel Disposal	HLW Disposal
Seaport Interim Storage Facility		
Years 1-9	---	3.4
Years 10-40	6.2	4.9
Years 41-49	6.2	3.4
Port Facility	1.5	1.5
Disposal and Monitoring Ships		
Years 1-9	---	14.5
Years 10-40	20.9	20.9
Years 41-49	20.9	14.3

total costs are estimated to be about $398 million for the spent fuel disposal and $721 million for the HLW disposal.

6.1.4.7 Safeguard Requirements

Because this concept may involve both subseabed and mined geologic disposal, its implementation could require safeguarding two separate disposal paths. The risk of diversion for the subseabed disposal concept would be primarily a short-term concern because of the remoteness of the disposal site and the major operational and equipment requirements that would have to be satisfied for retrieval. Physical protection of the sensitive facilities and transportation operations would be the most effective way to deny access to the waste for the short term, as is common to most waste disposal concepts. See Section 4.10 for additional discussion of predisposal operations safeguards requirements.

6.1.5 Ice Sheet Disposal

6.1.5.1 Concept Summary

It is estimated that, without significant climatic changes, the continental ice sheets could provide adequate isolation of high-level radioactive waste from the earth's biosphere. However, the long-term containment capabilities of ice sheets are uncertain. Areas of uncertainty have been reviewed by glaciologists (Philberth 1958, Zeller et al. 1973, and Philberth 1975). These reviewers cited the advantages of disposal in a cold, remote, internationally held area and in a medium that should isolate the wastes from man for many thousands of years to permit decay of the radioactive components. But they concluded that, before ice sheets can be considered for waste disposal applications, further investigation is needed on:

- Evolutionary processes in ice sheets
- Impact of future climatic changes on the stability and size of ice sheets.

Most of the analysis in these studies specifically addresses the emplacement of waste in either Antarctica or the Greenland ice cap. Neither site is currently available for waste disposal for U.S. programs: Antarctica because of international treaties and Greenland because it is Danish territory.

Proposals for ice sheet disposal suggest three emplacement concepts:

- Meltdown - emplaced in a shallow hole, the waste canister would melt its own way to the bottom of the ice sheet
- Anchored emplacement - similar to meltdown, but an anchored cable would allow retrieval of the canister
- Surface storage - storage facility would be supported above the ice sheet surface with eventual slow melting into the sheet.

Ice sheet disposal, regardless of the emplacement concept, would have the advantages of remoteness, low temperatures, and isolating effects of the ice. On the other hand, transportation and operational costs would be high, ice dynamics are uncertain, and adverse global climatic effects are a possibility.

6.1.5.2 System and Facility Description

Systems Options

The reference concept for the initial ice sheet disposal of nuclear waste has been developed from a number of options available at each step from the reactor to disposal in the ice sheet. It includes the three basic emplacement options and was selected through judgment of a "most likely" approach based on available information and is not supported by a detailed system engineering analysis.

Various options to be considered for ice sheet disposal are summarized in Figure 6.1.14. The bases for selection of the options chosen for the reference design (those blocked off) are detailed in a variety of source material cited in Appendix M.

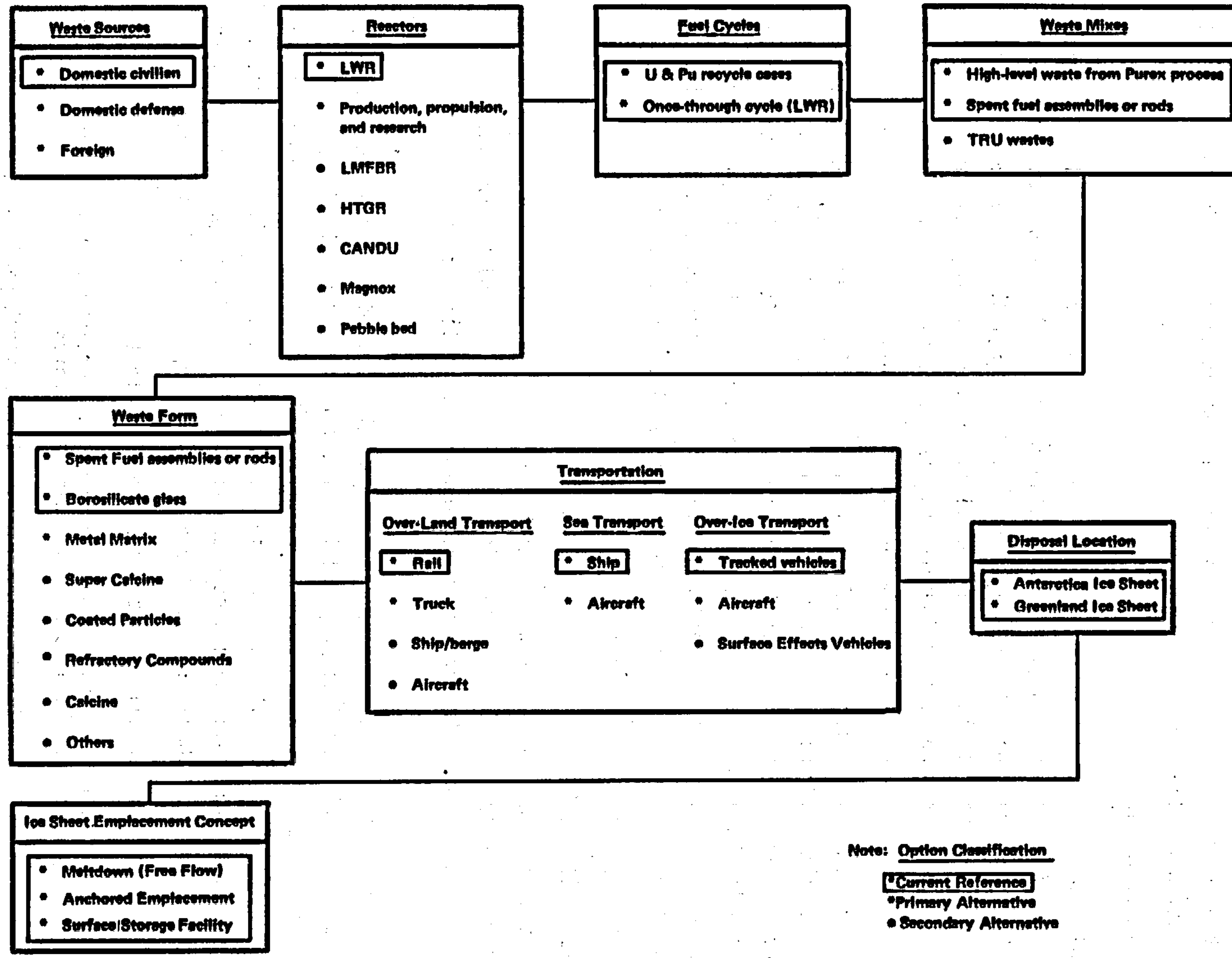


FIGURE 6.1.14. Major Options for Ice Sheet Disposal of Nuclear Waste

Because the options for the waste disposal steps from the reactor up to, but not including, the transportation alternatives are similar to those for a deep geologic repository, the options selected for the reference design are similar for the two concepts. From that point on, the options selected for the reference ice sheet design are based on current program documentation for ice sheet disposal.

Waste-Type Compatability

Ice sheet disposal by meltdown has been considered primarily for solidified, high-level wastes from nuclear fuels reprocessing. It would also be applicable for direct disposal of spent fuel, without reprocessing, although meltdown would be marginal if the fuel were emplaced 2 years after reactor discharge. The feasibility of meltdown emplacement of cladding hulls and fuel assembly hardware is questionable because the canister heating rate from radioactive decay would be less than 1/10 that in HLW waste canisters.

For most TRU waste, the heating rate would be less than 1/1000 that expected in HLW waste canisters, and the meltdown concept does not appear to be feasible. Without blending with HLW, disposal of this waste would be limited to storage in surface facilities on the ice or emplacement in shallow holes in the ice. For these options, the waste would be buried gradually in the ice sheet. Contact handled and remotely handled TRU wastes could be handled in a similar manner. Because of volume and cost considerations, TRU wastes are assumed to be placed in other terrestrial repositories.

Waste System Description

The ice sheet waste management system is detailed in Figure 6.1.15. This system concept is very similar to the very deep hole concept since both spent fuel and the uranium-plutonium recycle cases could be treated and mined geologic repositories could augment disposal.

The reference ice sheet disposal concept is not yet well defined. None of the three basic emplacement concept alternatives proposed in the literature (Battelle 1974, EPA 1979, and ERDA 1976) has been selected as a reference or preferred alternative. Waste disposal by any one of these three concepts would be either in the Antarctica or Greenland ice sheets. A generalized schematic of the waste management operational requirements is provided in Figure 6.1.16 (Battelle 1974). The schematic shows the basic system operations (EPA 1979):

- Predisposal treatment and packaging at the reprocessing plant
- Transporting solidified waste from the reprocessing plant or interim retrievable surface storage facility by truck, rail, or barge to embarkation ports
- Marine transport by specially designed ships during 1 to 3-month periods of each year.
- Unloading the waste canisters at a debarkation facility near the edge of the land mass
- Transporting over ice by special surface vehicles or aircraft on a year-round basis, as practicable
- Unloading and emplacing the waste canisters at the disposal site.

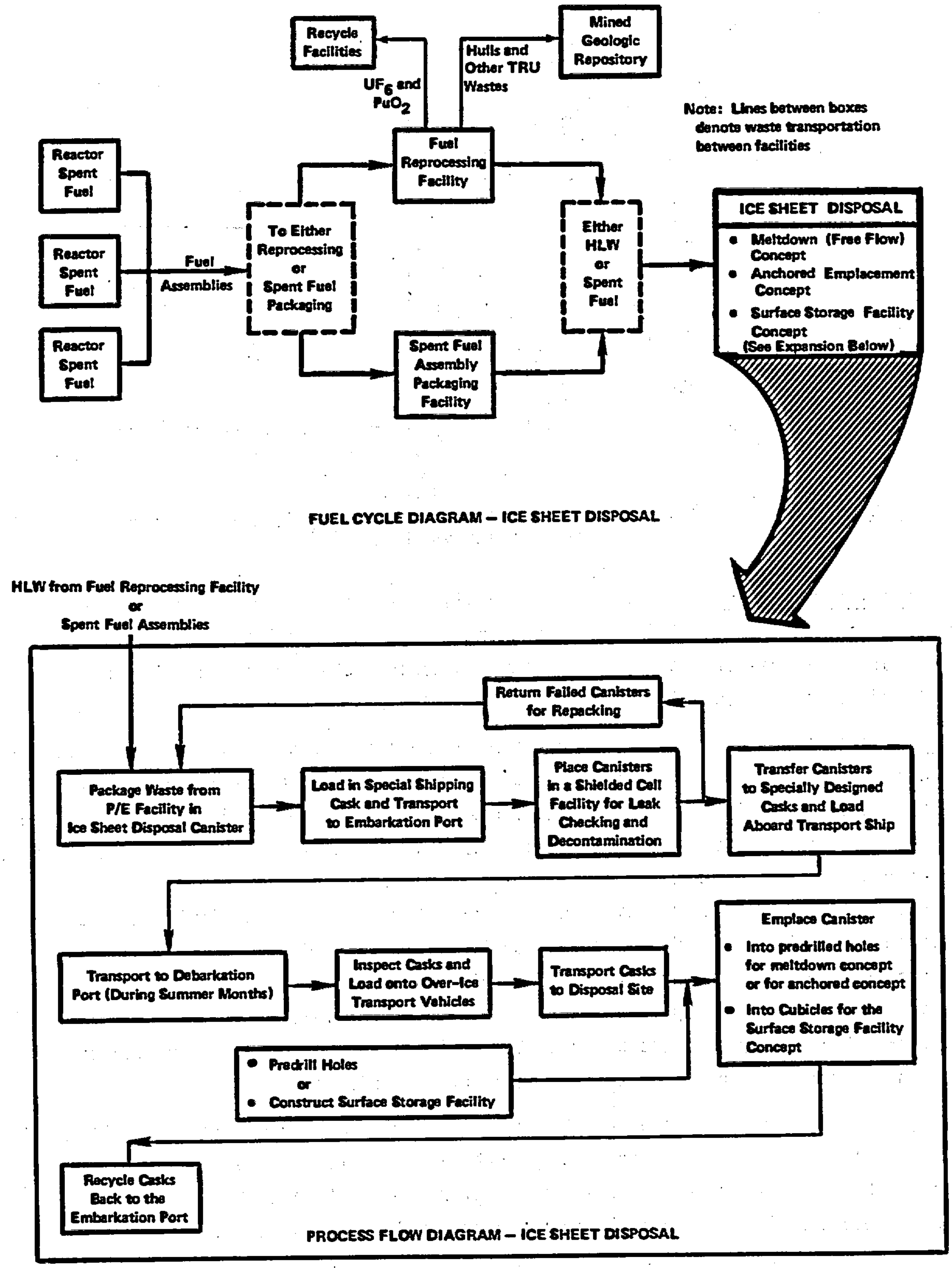


FIGURE 6.1.15. Waste Management System--Ice Sheet Disposal

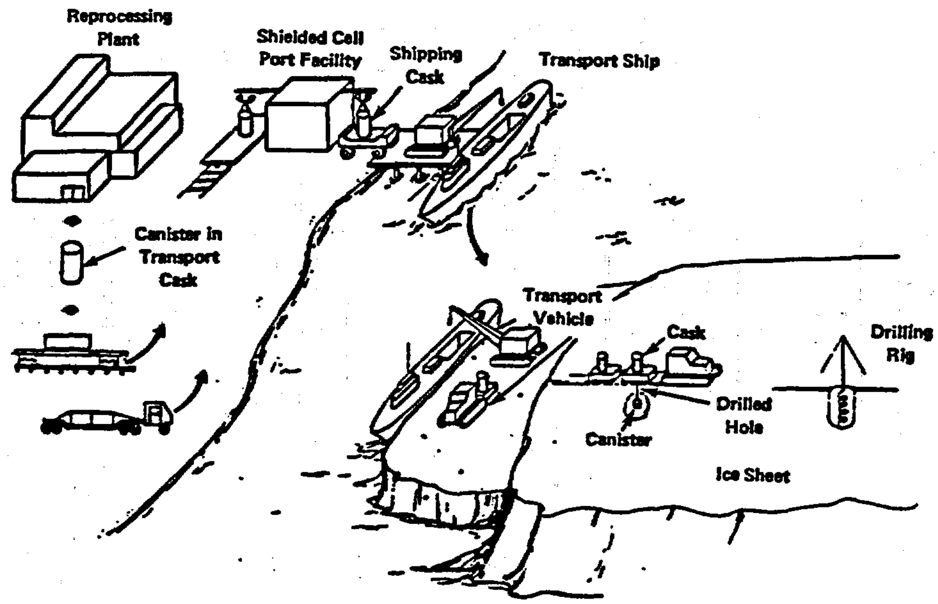


FIGURE 6.1.16. Schematic of Operations in Ice Sheet Disposal Systems for High-Level Radioactive Wastes(18)

Predisposal Treatment and Packaging. The predisposal treatment of waste for the ice sheet concept would be identical in many respects to the predisposal treatment of waste for the mined geologic repository concept. Chapter 4 discusses the predisposal systems for both spent fuel and HLW common to all the various alternative concepts for waste disposal.

Transportation and Handling. Transportation to the disposal site would probably be accomplished in three steps, as indicated above. First, all the waste canisters would be loaded into heavily shielded transport casks for shipment from the interim storage site to the embarkation port. Waste containers would accumulate at the embarkation port in the U.S. on a year-round schedule. There, the canisters would be unloaded in a shielded cell facility and examined for leakage, contamination, damage, or other unsuitable conditions. The canisters would be overpacked, transferred individually to specially designed casks, and loaded aboard a specially designed transport ship for shipment to the ice sheet. Acceptable canisters could also be stored for up to a year in an interim retrievable surface storage facility (Szulinski 1973). Any unacceptable canister would either be corrected on site or returned to the reprocessing plant or another appropriate handling facility.

Landing and discharge operations at the ice sheet would require special facilities and would be limited to the summer months. At the debarkation port, the casks would be inspected and unloaded onto over-ice transport vehicles. After transport to the disposal site, the canisters would be lowered from the casks to the emplacement site and the casks would be recycled back to the embarkation port. An alternative transportation mode would be to fly the waste canisters from the debarkation site to the emplacement site.

It appears possible, as an alternative, that the same shipping cask might be used for handling a waste canister first at the reprocessing plant, then for marine transport to the ice sheet, and finally for over-ice transport to the disposal site.

Debarkation ports on the ice sheets with handling systems for unloading casks directly onto the over-ice transport system would be possible in the Antarctic or in Greenland, but might be very expensive. The currently preferred alternative is to dock the transport ship at a land-based port in an ice-free area to unload the casks into the over-ice transport vehicles.

Emplacement. The waste canisters would be disposed of using one of the three basic concepts described in detail below.

The meltdown or free flow concept is shown in Figure 6.1.17 (ERDA 1976). Waste would be disposed of by selecting a suitable location in the ice sheets, predrilling a shallow hole, lowering the canister into the hole, and allowing it to melt down or free flow to the ice sheet base and bedrock beneath (EPA 1979).

The surface holes would be predrilled to depths from 50 to 100m and would provide protective shielding from radiation during canister emplacement. To avoid individual canisters interfering with each other during descent and possible concentration at the ice sheet base, the suggested spacing between holes is about 1000 m.

The canister meltdown rate is based on calculations from the penetration rates of thermal ice probes. It is estimated that the rate of descent for each canister would be on the order of 1.0 to 1.5 m/day. Assuming only vertical movement and an ice sheet 3000 m (9900 ft) thick, meltdown to the bedrock would take 5 to 10 years.

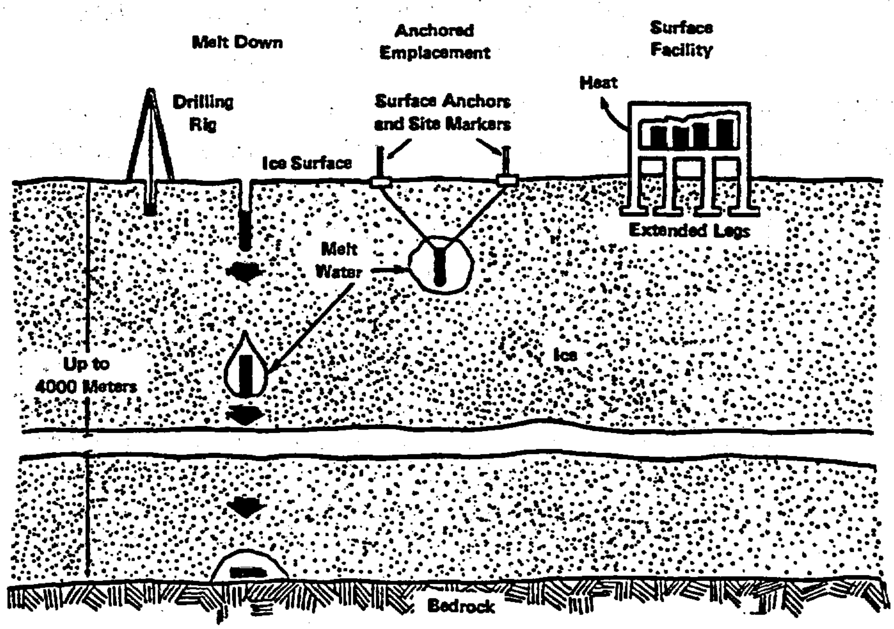


FIGURE 6.1.17. Ice Sheet Emplacement Concepts

An important factor in this concept would be the design and shape of the canister, which should help assure a vertical path from surface to bedrock. In addition to the canister design and shape, the type of construction materials would be important. Specifications for these materials would have to include consideration of differences in ice sheet pressure and the possibility of saline water at the ice/ground interface. A multibarrier approach that gives consideration to the total waste package and its emplacement environment would be required. This approach would be equally applicable to the anchored emplacement and surface storage alternatives.

The anchored emplacement concept, also shown in Figure 6.1.17, would require technology similar to that required by the meltdown or free flow concept described above, the difference being that this concept would allow for interim retrieval of the waste (EPA 1979). Here, cables 200 to 500 m (660 to 1650 ft) long would be attached to the canister before lowering it into the ice sheet. After emplacement the canister would be anchored at a depth corresponding to cable length by anchor plates on or near the surface. The advantage over the meltdown concept is that instrument leads attached to the lead cable could be used to monitor the condition of the canister after emplacement.

Following emplacement, new snow and ice accumulating on the surface would eventually cover the anchor markers and present difficulties in recovery of the canister. The average height of snow and ice accumulating in the Antarctic and Greenland is about 5 to 10 cm/yr (2 to 4 in./yr) and 20 cm/yr (8 in./yr), respectively. However, climatic changes might result in a reversal of this accumulation with ice being removed from the surface by erosion or sublimation. If continued for a long period of time such ice surface losses could expose the wastes. Recovery of canisters 200 to 400 years after emplacement might be possible by using 20-m (66-ft)-high anchor markers. It would take about 30,000 years for the entire system to reach ice/ground interface at a typical site. During that time, the canisters and anchors would tend to follow the flow pattern of the ice (Battelle 1974).

The surface storage facility concept would require the use of large storage units constructed above the snow surface (EPA 1979). The facilities would be supported by jack-up pilings or piers resting on load-bearing plates, as shown in Figure 6.1.17. The waste canisters would be placed in cubicles inside the facility and cooled by natural draft air. The facility would be elevated above the ice surface for as long as possible to reduce snow drifting and heat dissipation. During this period, the waste canisters would be retrievable. However, when the limit of the jack-up pilings was reached, the entire facility would act as a heat source and begin to melt down through the ice sheet. It is estimated that such a facility could be maintained above the ice for a maximum of 400 years after construction (Battelle 1974).

Retrievability/Recoverability. Waste disposed of using the meltdown emplacement concept would be retrievable for a short period, but movement down into the ice and successful deployment of the concept design would quickly render the waste essentially irretrievable. Recovery is also considered nearly impossible. Retrievability for the other two emplacement concepts is indicated in the discussions above.

6.1.5.3 Status of Technical Development and R&D Needs

Present State of Development

Ice sheet disposal is in the conceptual stage of development and an extensive R&D program would be required to implement an operational disposal system (EPA 1979 and DOE 1979). Current technology appears adequate for initial waste canister emplacement using the concepts described. Necessary transportation and logistics support systems could be made available with additional R&D. The capability of ice sheets to contain radioactive waste for long periods of time is at present only speculative, because of limited knowledge of ice sheet stability and physical properties. Verification of theories that support ice sheet disposal would require many years of extensive new data collection and evaluation.

Technological Issues to be Resolved

Key technical issues that would have to be resolved for development of the ice sheet disposal concept include:

Choice of Waste Form

- Behavior of glass or other waste forms under polar conditions
- Ability of container to withstand mechanical forces.

Design of Shipping System for Polar Seas

- Extremes of weather and environmental conditions expected
- Debarkation port design
- Ship design
- Cask design
- Recovery system for cask lost at sea.

Design of Over-Ice Transport

- Crevasse detector
- Navigational aids
- Ability to traverse surface irregularities, snow dunes, and steep ice slopes
- Maintenance of road systems
- Recovery system for lost casks.

Design of Monitoring for Emplaced Waste

- Location, integrity, and movement of emplaced canisters

- Radioactivity of water at ice-rock interface
- Hydrologic connections to open oceans and effects on ice stability.

In addition, there are serious issues connected with the ability to adequately predict long-term ice sheet behavior, including rates of motion within the sheet, the physical state and rates of ice flow, movement of meltwater at the base of the sheet, and the long-term stability of the total sheet.

R&D Requirements to Make System Operational

R&D requirements to resolve these issues may be grouped in terms of those related to the handling, transportation, and emplacement of the waste, and those related to obtaining basic information on ice sheets. In the former group, R&D would be required in the areas of waste forms (content, shape, and materials), transportation (shielding, casks, ships, aircraft, over-ice vehicles), facilities (port, handling, inspection, repair), and supply logistics (fuel, equipment, personnel requirements). Research needs applying to ice sheets would include determination of ice sheet movement and stability through geological/geophysical exploration and ice movement measurements, studies of ice flow mechanics including effects of bottom water layers, studies of global and polar climatology, and acquisition and analysis of meteorological and environmental data.

Estimated Implementation Time and R&D Costs

If the ice sheet disposal concept were to prove viable, the time required to achieve an operating system is estimated to be about 30 years after the start of the necessary research program. The research program itself would require about 15 years of activity directed primarily toward improved understanding of ice sheet conditions, selecting an emplacement method, identifying and assessing ice sheet areas most suitable for the method selected, and research and preliminary development of systems unique to the particular emplacement method and site. Should the research program culminate in a decision to proceed with project development, an additional period of 12 to 13 years would be required to implement an operational disposal system.

R&D costs for ice sheet disposal are estimated to be $340 million (in 1978 dollars) for the initial research and preliminary development program and between $570 million and $800 million for development, depending on the emplacement mode chosen.

Summary

Major uncertainties, shortcomings, and advantages of the concept are summarized below:

- The environment involved is non-benign to men and equipment, and the transportation limitations are severe.
- Understanding and performance assessment of the subsurface mechanisms of transport and package degradation are not developed to any degree.
- The concept does have the capacity for multiple barriers.

- The capability for corrective action over a long period is uncertain, and site selection criteria and performance assessment capability are nonexistant.
- No site is currently, or potentially in the future, available to the U.S. for R&D.

6.1.5.4 Impacts of Construction and Operation (Preemplacement)

Health impacts, both radiological and nonradiological, and natural system impacts are analyzed below.

Health Impacts

Radiological impacts would in many ways be similar to those for mined geologic disposal but would have the added problem of extensive interim storage. Nonradiologic impacts might occur both as a result of routine operations or in abnormal or accidental conditions.

Radiological Impacts. Ice sheet disposal would be different from the mined geologic repository and other alternatives because of the requirement for extensive interim storage of either processed waste or spent fuel. Such storage would be necessary because lead times for research, development, and testing are 10 to 30 years longer than those for geologic disposal (DOE 1979). During this time, radiological effects would include doses to occupational personnel, the normal release of radioactive effluents to the atmosphere, and the potential for accidental release of radioactivity. At this time, no studies are available that provide a quantitative estimate of these impacts; however, it is expected that they would be similar to those from fuel storage facilities.

Preparation of waste for ice sheet disposal would be similar to that for mined geologic disposal methods. Likewise, the radiological effects associated with this option are assumed to be similar to those associated with geologic disposal methods. The radiological risks and impacts from the transportation of the waste would be to the Artic or Antarctic essentially the same as those discussed in subseabed disposal. The ice sheet disposal option is not sufficiently developed to estimate the radiological effects of routine operations on the ice sheet.

Accidents while unloading at the ice shelf seaport or during transport over the ice could create retrieval situations that would be difficult in the polar environment. Quantitative estimates of the radiological impact of such accidents are not available.

Nonradiological Impacts to Man and Environment. Potential nonradiological impacts could occur during all phases of ice sheet disposal operations. As with many of the alternative disposal strategies, impacts can be categorized as to whether they would occur during waste preparation, transportation, or emplacement activities. In general, those impacts associated with transportation and emplacement would warrant the most analysis. Waste preparation impacts would be similar to those for other disposal strategies discussed earlier.

Occupational casualties from the nonpolar activities are expected to occur at rates typical of the industrial activities that would be involved, and to be independent of both the nuclear and polar aspects of the remainder of the system. Operations are routinely carried out with nuclear systems and in the polar regions with safety comparable to that experienced in more familar environments. In all likelihood, the required large-scale activities could also be performed safely, with the polar conditions being reflected in higher program costs rather than in decreased safety.

Accidents in processing and handling the waste material could occur before the material reaches the embarkation facility. Impacts resulting from such accidents are common to virtually all of the alternative disposal options. Other impacts would be virtually identical to those of the subseabed disposal option because in both cases the material would be transported to a coastal location.

Nonradiological health effects for activities that would occur on the ice sheet under abnormal conditions have not been studied extensively. Occupational impacts would occur, but as stated above, it is not expected that polar conditions will significantly alter the level of effects anticipated. Non-occupational effects would be even less significant, reflecting the lack of human activity on the ice sheets.

Natural System Impacts

Quantitative estimates of the radiological impact of ice sheet disposal on the ecosystem are not available. These impacts are expected to be small because there are very few living organisms in the polar regions, except along the coastline. Nonradiological ecological impacts at the disposal site are difficult to characterize because of a lack of understanding of the processes occurring in polar environments. The present understanding of impacts on the glacial ice mass or the dry barren valleys of Antarctica is limited. The effect of the heat that would be produced by the wastes on the ice or the potential geologic host media remains unclear.

Air impacts would result from the combustion products of over-ice transport vehicles, support aircraft, and fuel consumed for heating the facilities at the various sites. At present, the effects of these products are not considered a major problem.

Few, if any, ecological impacts are expected near the disposal sites because the plant and animal life are confined mostly to the coastal areas. Access routes and air traffic lanes could be made to avoid as much as possible the feeding, nesting, and mating spots of the birds and animals that inhabit the coastal areas. Fuel spills, equipment emissions, and general transportation support activities could lead to some localized impacts along the transportation disposal corridors. Few, if any, other impacts on water are expected, except for a marginal increase in temperature of the water that would be used for once-through cooling of canisters during sea transport. The only other water uses would be for consumption by the 200 operating personnel, which would be obtained by melting the ice.

Other possible land impacts considered in the reference study include accidental spills of fuel and the probability of fuel bladders rupturing during drop-offs. Rupture of the fuel bladders is considered to be a high risk because the fuel is capable of penetrating the snow and could reach the underlying ice where it would remain until evaporated or eventually buried by additional snow. Accidental spills could reach the ocean if the incident occurred near the edge of the ice sheet.

Socioeconomic Impacts

Socioeconomic impacts for the ice sheet disposal option would be similar to those for the island and subseabed disposal options. Because these options are still at the concept level, however, detailed socioeconomic assessments are not possible. In general, socioeconomic impacts would be experienced where handling facilities are constructed and operated.

Impacts that might be expected where handling facilities would be constructed include disruptions or dislocations of residences or businesses; physical or public-access impacts on historic, cultural, and natural features; impacts on public services such as education, utilities, road systems, recreation, and health and safety; increased tax revenues in jurisdictions where facilities would be located; increased local expenditures for services and materials; and social stresses.

The operating work force required for a dock facility would likely be comparable to that for any moderate-size manufacturing facility and impacts would vary with location. Impacts would be primarily in housing, education, and transportation, with no significant impacts on municipal services. Impact costs would presumably be offset by revenues, but socioeconomic considerations at this stage are not easily quantified.

Aesthetic Impacts

Aesthetic impacts are expected to be insignificant because of the remoteness of the area and the lack of permanent residence population (EPA 1979).

Aesthetic impacts for the ocean transportation activities and embarkation facilities would be very limited and similar to those of subseabed disposal. The waste packaging and transportation activities that would be a part of the ice sheet disposal process would have aesthetic impacts similar to those of mined geologic repositories. Noise, fugitive emissions, and the appearance of facilities and equipment used to prepare and transport the waste material are common to a number of disposal options. These impacts are generally reviewed in Chapter 4.

Resource Consumption

Predisposal activities would include packaging and transportation of spent fuel to seaports for shipment to the receiving port at the ice sheet, if spent fuel were disposed of rather than reprocessed waste. If reprocessing of spent fuel were undertaken, then predisposal activities would also include conversion of the waste to a high-integrity form, like glass, before transportation to seaports. The resource requirements of these activities have been discussed elsewhere in this document for other disposal alternatives, and would be the same for ice sheet disposal, except for differences in transportation routings.

Little quantitatives information exists on the energy, resource, and land requirements unique to ice sheet disposal. Ice sheet disposal would require construction of ships, airplanes, and over-the-ice vehicles that would not be required for other disposal alternatives. A greater number of shipping casks would also be required, because of the long cask turn-around time.

Transporting the waste material to its final destination across the ice fields would also require expenditure of energy. Either surface or air transport would use large quantities of fuel because of the great distances involved.

Some land impacts would probably be experienced in connection with the embarkation port facility. An area of about 1km^2 (0.4 mi^2) would be required for the shielded cell and the loading dock facilities. The port facility would be equipped with its own separate water, power, and sewer systems to assure maximum safety. The over-ice transport routes would include an area at the edge of the ice sheet, ice shelf-edge, and ice-free areas on land for unloading the shipping casks. Approximately six support and fueling stations would be required along the transport route to the disposal area. Land requirements at the disposal site are estimated at 11,000 km^2 (4,2000 mi^2) for waste from a plant producing 5 MTHM/day based on a waste canister spacing of one/Km.

International and Domestic Legal and Institutional Considerations

The ice sheet disposal option, like the island and subseabed options, would require transporting waste material over the ocean, and the general international implications of such transportation are important.

Numerous legal and institutional considerations would emerge if the ice sheet disposal concept were seriously pursued in either Greenland or Antartica. In the case of Greenland, treaty arrangements would have to be made with Denmark because Greenland is a Danish Territory.

In the case of Antarctica, a number of treaties and agreements exist that could affect the use of the ice sheets for storage and disposal of radioactive material. Disposal of waste in Antarctica is specifically prohibited by the Antarctic Treaty of 1959, of which the United States is a signatory (Battelle 1974). The treaty may be renewed after it has been in effect for 30 years, or amended at any time.

Outcomes of two meetings reflect the current range of international attitudes toward ice sheet waste disposal. One attitude was expressed in a resolution passed by the National Academy of Sciences, Committee on Polar Research, Panel on Glaciology, at a meeting in Seattle, Washington, May, 1973. The resolution neither favored nor opposed ice sheet waste disposal as such. However, a statement from a second meeting, on September 25, 1974, in Cambridge, England, attended by scientists from Argentina, Australia, Japan, Norway, the United Kingdom, the United States, and the USSR, recommended that the Antarctic ice sheet not be used for waste disposal.

6.1.5.5 Potential Impacts over the Long Term (Postemplacement)

Potential Events

Long-term impacts with the greatest potential significance are related to glacial phenomena that are not well understood. For example, ice dynamics and climatic variations affecting glaciation might be altered by waste disposal activities. Regardless of whether meltdown, anchored emplacement, or surface storage were used, potentially major modifications in the delicately balanced glacial environment could occur.

One of the major areas of uncertainty stems from our limited understanding of ice sheet conditions. Little is known of the motion of the continental ice sheets except for surface measurements made close to the coast (Gow et al. 1968). Three general types of flow have been defined--sheet flow, stream flow, and ice-shelf movement (Mellor 1959). Each type of flow appears to possess a characteristic velocity. It is also believed that ice sheets where bottom melting conditions exist may move almost as a rigid block, by sliding over the bedrock. Where there is no water at the ice-bedrock interface, it is believed that the ice sheet moves by shear displacement in a relative thin basal layer. The formation of large bodies of water from the waste heat could affect the equilibrium of such ice sheets.

In addition, two potential problems concerning the movement of the waste are unique to an ice sheet repository. First, the waste container would probably be crushed and breached once it reached the ice/ground interface as a result of ice/ground interaction. Second, the waste might be transported to the sea by ice movement.

Compared with other disposal schemes, the probability of human intrusion would be very low because the disposal area would be located in the most remote and inaccessible part of the world, presently with a low priority for exploration of natural resources or habitation. The lack of human activity in these areas would markedly decrease the chance of humans disturbing waste material emplaced in an ice sheet. Conversely, because of the remoteness of these areas they are relatively unexplored. Therefore they could attract considerable future resource exploration.

Potential Impacts

After the waste is emplaced and man's control is relinquished or lost, possible impacts fall into two broad categories. One of these relates to the reappearance of the radioactive waste in the environment, and the other involves the chance that the presence of waste would trigger changes in the ice sheets that would have worldwide consequences. For options that would place the waste within the ice or at the ice/ground interface, significant research would be required to predict future ice movements, accumulation or depletion rates, subsurface water flow rates, frictional effects at the interface, and trigger mechanisms. A major purpose of this research would be to compare the degree of sensitivity of the predicted behavior to man's ability to forecast long-term situations such as global weather patterns, stability of the ice sheets, and sea-level changes.

Specific areas of concern, as discussed below, are:

- Effects of waste on ice sheet environment
- Effects of ice sheet on waste
- Effects of waste on land environment.

Effects of Waste on Ice Sheet Environment. If waste canisters were allowed to reach or approach the bottom of the ice, they could possibly generate sufficient heat to produce a water layer over a large portion of the bottom surface of the ice. Furthermore, melt pools around the canisters could conceivably coalesce and also unite with any subglacial water, in the disposal area, to form a large water mass within the ice or at the edge of the ice-bedrock interface. Either event might trigger an increase in the velocity of the ice mass and perhaps produce surging. It has been postulated that major surges in the East Antarctica ice sheet could affect solar reflection and alter the sea level. The most extreme effect would be the start of glaciation in the Northern Hemisphere (Wilson 1964). The accelerated movement could also move emplaced material toward the edge of the ice sheet, possibly reducing the residence time. Basal ice sheet water could also conceivably form a pathway for transporting waste material from the disposal area to the edge of the ice sheet, and thus to the ocean.

Hypothetical dose calculations have been made for radionuclides released from an ice sheet disposal site into the ocean off the coast of Greenland (EPA 1979). On the basis of assumptions that a failure occurs in the disposal system, the release of radionuclides into the Greenland current of 8×10^6 m^3/sec would be 0.3 percent/yr of the total inventory available. Complete mixing could occur in the ocean. Human pathways are assumed to be mostly via fish consumption. The maximum dose was considered to be from an individual consuming 100 kg/yr of fish caught in these contaminated waters and is estimated to be 0.2 mrem/yr. Further discussion of radioactive releases to the ocean is included in Section 6.1.4.5 on the subseabed concept.

Effects of Ice Sheet on Waste. Movement of the ice sheet might cause shearing or crushing of canisters, allowing water to come in contact with the waste form so that leaching could occur. Such breakage would most likely occur when the canisters are moved along the ice-bedrock interface.

If major climatic changes were to produce an increase in temperature in the polar region, the ice sheet might erode to such an extent that it would allow the waste to be much closer to the edge of the ice. The temperature increase could also increase the velocity of the ice movement toward the coast.

Effects of Waste on Land Environment. As in the case of space and subseabed disposal, geologic repository facilities are assumed to be constructed for TRU and other wastes not disposed of through the procedures established for the majority of HLW. Long-term effects could result from these auxiliary activities. These impacts would be similar to those described for the mined geologic concept. The other land area that could be impacted is the region of dry barren valleys in Antarctica. If wastes were placed in this area, impacts would be very similar to those of the mined geologic repository. The major difference would be that the ground-water regime in Antarctica would mostly affect remote frozen ground-water systems.

Terrestrial ecosystems in the ice sheet regions under study for disposal sites are limited in diversity. Severe climatic conditions limit most organisms to the seaward margins of both Greenland and Antarctica. Consequently, the potential for impact to terrestrial organisms in the ice sheet disposal is quite limited. Potentially more significant are the long-term ecological effects of any accidents that would occur on the land mass where the wastes were generated. As described in Section 5.6, these impacts should not be significant unless an accident or encroachment occurs.

6.1.5.6 Cost Analysis

The cost of depositing nuclear wastes in ice sheets is currently expected to be relatively high; higher, for example, than the cost of geologic emplacement in the U.S. This is primarily because of the high costs for R&D as presented in Section 6.1.5.3. Capital, operating, and decommissioning cost estimates are presented below.

Projected Capital Costs

Projected capital costs for ice sheet emplacement of 3000 MT/yr of spent fuel, or the wastes recovered from processing that amount of fuel, are $1.4 billion to $2.3 billion as shown in Table 6.1.17.

Projected Operating Costs

Projected operating costs for the emplacement of 3000 MT/yr of spent fuel or HLW are shown in Table 6.1.18.

Decommissioning Costs

Decommissioning costs associated with contaminated equipment would probably be limited primarily to the shipping casks used to transport waste canisters for ice sheet disposal. These costs are estimated at $9.7 million, which is 10 percent of the initial capital cost of the shipping casks. Costs for decommissioning other facilities and equipment are assumed to be similar to those for other waste disposal alternatives.

6.1.5.7 Safeguard Requirements

Because the reference concept uses both ice sheet and mined geologic disposal, its implementation would require safeguarding two separate disposal paths. The risk of diversion for the meltdown concept would be basically a short-term concern because once the waste had been successfully disposed of in accordance with design, it would be considered irretrievable. For the anchored and surface storage concepts, although the waste would be considered retrievable for as long as 400 years, the harsh environment in which it would be

TABLE 6.1.17. Capital Costs For Ice Sheet Disposal (Millions of 1978 Dollars)

Case I. Meltdown or Anchored Emplacement: Surface Transportation

1.	Construction of Port Facilities	730
2.	Sea Transport Vessels	290
3.	Ice Breakers	190
4.	Over-Ice Transport Vehicles	100
5.	Drilling Rigs	50
6.	Monitoring Equipment	50
7.	Shipping Casks	100
8.	Aircraft	100
9.	Support Facilities	150
		1760

Case II. Surface Storage

1.	Construction of Port Facilities	730
2.	Sea Transport Vessels	290
3.	Ice Breakers	190
4.	Over-Ice Transport Vehicles	100
5.	Surface Storage Facility	500
6.	Monitoring Equipment	50
7.	Shipping Cask	100
8.	Aircraft	100
9.	Support Facilities	190
		2250

Case III. Meltdown or Anchored Emplacement: Aerial Emplacement

1.	Construction of Port Facilities	500
2.	Sea Transport Vessels	150
3.	Aircraft	500
4.	Shipping Casks	100
5.	Monitoring Equipment	50
6.	Support Facilities	150
		1450

TABLE 6.1.18. Operating Costs For Ice Sheet Disposal
(Millions of 1978 Dollars/Year)

Emplacement Concept	Meltdown or Anchored		Surface Storage
Emplacement Method	Surface	Aerial	Surface
Cost Category:			
Operating Personnel[a]	34	29	39
Material & Consumables[b]	58	29	58
Services & Overhead[c]	68	58	78
Capital Recovery[d]	175	141	224
Total	335	257	399

(a) Based on \$50,000/man-year.
(b) Including \$29 million/yr and \$5 million/yr port upkeep for surface and aerial emplacement, respectively.
(c) Based on twice the operating personnel costs.
(d) Based on 10 percent of capital expenditures (not including research and development costs). Encapsulation costs not included.

placed and the equipment needed for retrieval would also make any risk of diversion primarily a short-term concern. Only minimum safeguards would be required after emplacement. Physical protection of the sensitive facilities and transportation operations would be the most effective way to deny access to the waste for the short term, as is common to most waste disposal alternatives. See Section 4.10 for additional discussion of predisposal operation safeguard requirements.

6.1.6 Well Injection

6.1.6.1 Concept Summary

Well injection technology was initially developed by the oil industry for the disposal of oil field brines. These brines were usually pumped back into the original reservoir and, in some cases, used to "drive" the oil toward a producing well. The well injection concept has subsequently been used for the disposal of various natural and industrial wastes. The techniques developed in the oil industry handle liquid wastes only - particulate matter can cause blocking of the pores in rock.

A well injection process using grout was developed by Oak Ridge National Laboratory (ORNL) for the injection of remotely handled TRU liquid radioactive wastes into shale strata (ERDA 1977). This technique is also suitable for grout slurry wastes, and a new facility is now under construction at ORNL for liquid and slurry waste injection (ERDA 1977). Well injection could be a low cost alternative to deploy and operate because of the widespread use of the required techniques and the "off-the-shelf" availability of the main components. Two reference methods of well injection are considered in this section: deep well liquid injection and shale grout injection.

Deep well injection would involve pumping acidic liquid waste to depths of 1,000 to 5,000 m (3,300 to 16,000 ft) into porous or fractured strata suitably isolated from the biosphere by overlying strata that are relatively impermeable. The waste may remain in liquid form and might progressively disperse and diffuse throughout the host rock. This mobility within the porous host media formation might be of concern regarding release to the biosphere. Questions have also arisen regarding the possibility of subsequent reconcentration of certain radioisotopes because of their mobility. This could lead to the remote possibility of criticality if, for instance, the plutonium is reconcentrated sufficiently. Isolation from the biosphere would be achieved by negligible ground-water movement in the disposal formation, particularly towards the surface, retention of nuclides due to sorption onto the host rock mineral skeleton, and low probability of breeching by natural or man-made events. The concept is not amenable to a multiplicity of engineered barriers.

For shale grout injection, the shale would first be fractured by high-pressure water injection and then the waste, mixed with cement and clays, would be injected into suitable shale formations at depths of 300 to 500 m (1,000 to 1,600 ft) and allowed to solidify in place in layers of thin solid disks. The shale has very low permeability and probably good sorption properties. The injection formations selected would be those in which it could be shown that fractures would be created parallel to the bedding planes and would therefore remain within the host shale bed. This requirement is expected to limit the injection depths to the range stated above. Direct operating experience is available at ORNL for disposal of TRU wastes by shale grout injection. The grout mixes have been designed to be leach resistant and hence the concept minimizes the mobility of the incorporated radioactive wastes.

Isolation from the biosphere is achieved by low leach rates of radionuclides from the hardened grout sheet, negligible ground-water flow particularly up through the shale strata, retardation of nuclide movement by minerals within the shale strata, and low probability of breeching by natural or man-made events.

6.1.6.2 System and Facility Description

System Options

The two reference concepts for well injection disposal of nuclear waste have been selected from a number of options available at each step from the reactor to disposal at the well injection facility. These two concepts are judged as "most likely" based on the status of current technology. A summary of various options to be considered for well injection disposal is illustrated in Figure 6.1.18. Additional pertinent data available on the options can be found in various source material listed in Appendix M.

Waste-Type Compatibility

For both reference concepts the waste form injected would be HLW. Since disassembly and some processing would be necessary for well injection, the concepts would be suitable for fuel cycles that recycle uranium and plutonium. However, well injection could also be applied to once-through fuel cycles after dissolution or slurrying of spent fuels. In these

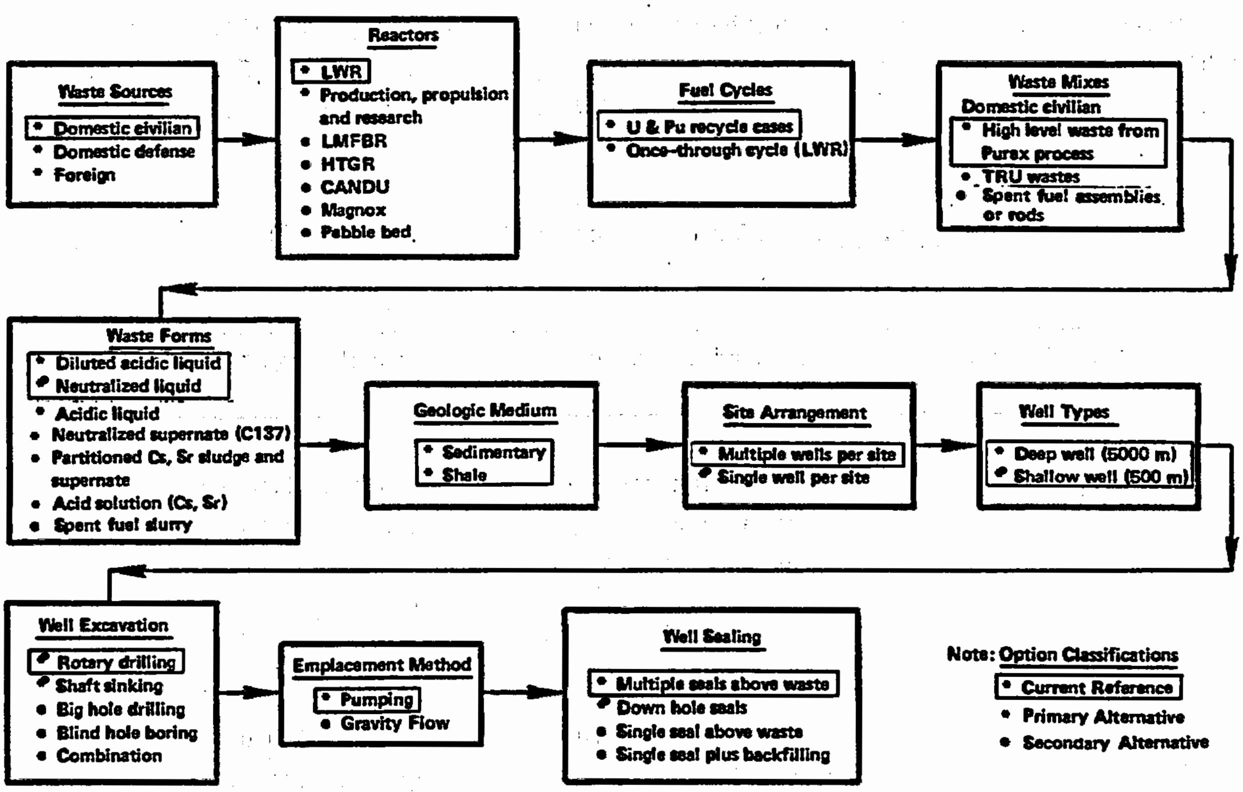


FIGURE 6.1.18. Major Options for Well Injection Disposal of Nuclear Waste

cases, the injection liquid would contain large amounts of actinides, which might affect the thermal properties and interaction mechanisms of the waste in the host media. Well injection might also be used to dispose of high-heat-level partitioned wastes, which could relieve high thermal loadings in a mined geologic repository for example. Note that retrieval would be difficult and incomplete using either concept, although deep well injection would have more potential for at least partial retrieval than would the shale-grout method, which would fix the waste in a relatively insoluble solid.

For deep well injection, the liquid waste would have to be substantially free from all solid matter to prevent clogging of the formation pores. Filtration down to 0.5 m particles is typical for process waste injection systems (Hartman 1968). The waste would have to remain acidic to ensure that all the waste products stay in solution.

For shale grout injection, neutralized waste (sludge and supernate) would be mixed with cement, clay, and other additives.

Waste System Description

The fuel cycle and process flows associated with the two reference options are illustrated on Figure 6.1.19. Significant features of these concepts are summarized in Table 6.1.19.

Both concepts are based on restricting the maximum temperature in the injection formation to 100 C (212 F), assuming a geothermal gradient of 15 C/km (44 F/mile), to avoid undesirable mineralogical effects that would occur at higher temperatures. (For example, comparatively large amounts of waste would be released from the clay mineral montmorillonite if

TABLE 6.1.19. Reference Concepts Summary (DOE 1979)

Reference Concepts	Depth of Injection	Disposal Formation
Deep well liquid injection	100-m-thick zone at average depth of 1,000 m	Sandstone with shale caprock at 950-m depth; porosity 10 percent
Shale grout injection	100-m-thick zone at average depth of 500 m	Shale extending to within 50 m of ground surface

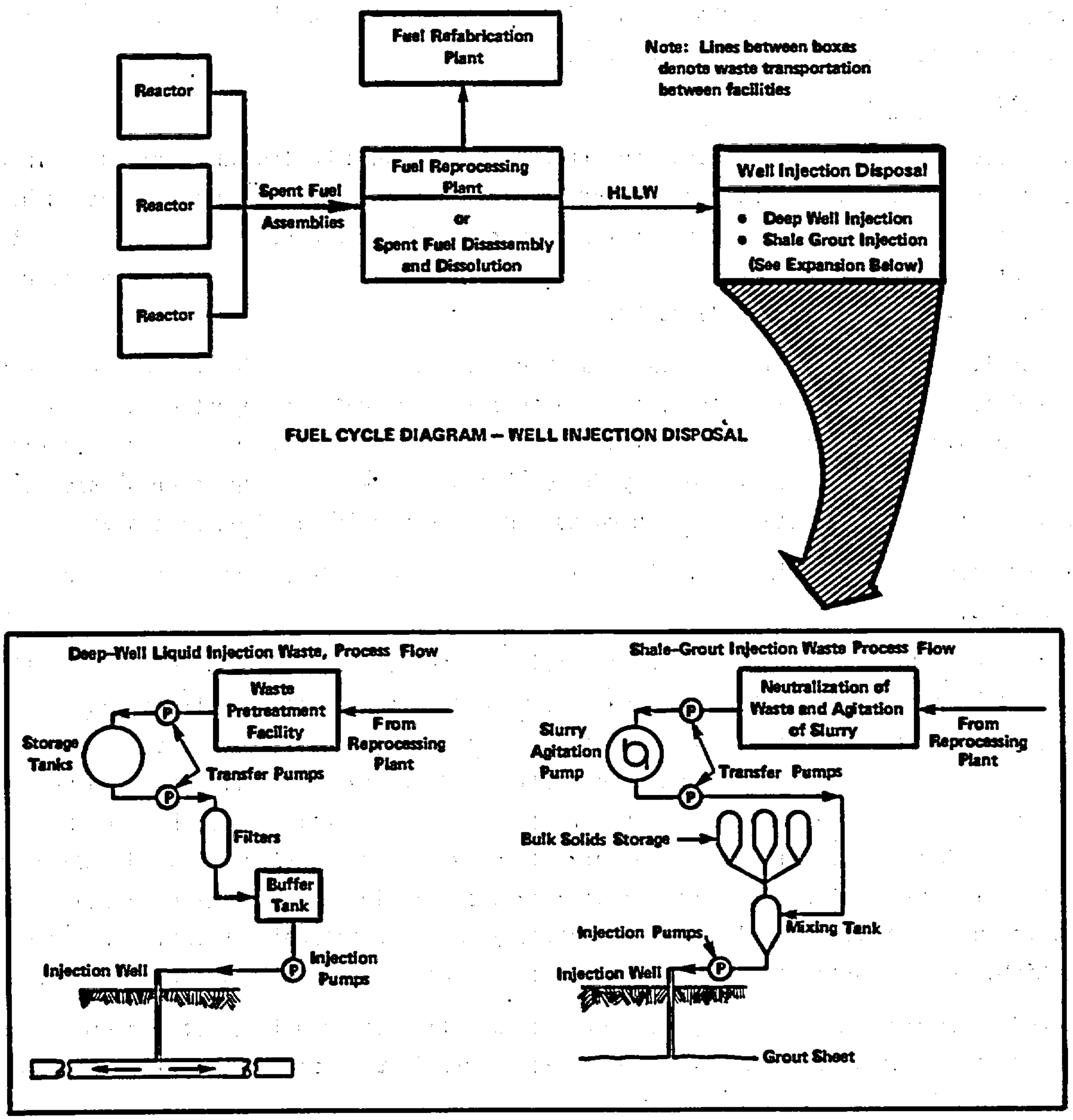


FIGURE 6.1.19. Waste Management System--Well Injection Disposal

heated to above 100 C) (EPA 1973). Although disposal strata containing more inert minerals, particularly quartz-rich sandstones suitable for deep well liquid injection, might sustain higher temperatures, thermal effects on containment formations, which may include temperature sensitive minerals, would also have to be considered.

Deep Well Injection

In the deep well injection concept, the liquid wastes would be fed into porous or fractured strata, such as depleted hydrocarbon reservoirs, natural porous strata, or zones of natural or induced fractures. To protect freshwater aquifers from waste contamination, the injection zone would have to be well below the aquifers and isolated by relatively impermeable strata, e.g., shales or salt deposits.

In general, injection requires pressure at the wellhead, although in some circumstances gravity feed is sufficient. The controlling factors are the rate of injection and the permeability of the disposal formation. The increase in the total fluid volume in an injection zone is accommodated by compression of any fluid already present and expansion of the rock formation. The relation between injection rates and pressures is based on extensive oil-well and ground-water experience. Injection is possible at depths down to several thousand meters.

For this concept, the activity of the injection waste has been assumed to be controlled by the allowable gross thermal loading, the injection zone thickness, and the porosity in that zone. It is also assumed that one injection zone with two wells would be used at each site. In the long term, the waste might progressively disperse and diffuse throughout the host rock and eventually encompass a large volume. The concentration might be variable and unpredictable. Thus, criteria for permissible activity levels might be required. Determination of the dilution requirement is complicated by the sorption of nuclides onto the mineral skeleton, to an extent determined by waste chemistry and rock mineral content. If sorption were too high, concentration of heat-generating components might result in "hot spots".

Injected waste might be partially retrieved by drilling and pumping, but sorption of nuclides onto the mineral skeleton and precipitation within the pores would limit the amounts recovered.

Predisposal Treatment. In deep well injection, spent fuel would be shipped to a processing facility at the well injection site. The spent fuel would be dissolved in acid and the hulls removed. (For recycle, the uranium and plutonium would be removed from the acid solution.) The acid solution would constitute the basic waste form for isolation.

The acid waste from reprocessing would contain both fission products and actinides. Between 60 and 75 percent of the heat generated in the initial emplacement years would be due to ^{90}SR and ^{137}Cs. Partitioning strontium and cesium from the remainder of the waste would permit different isolation practices to be adopted for the high-heat-generating, relatively short-lived isotopes (half-lives about 30 years) and the remainder of the waste containing the much longer lived, lower heat generating isotopes.

The liquid waste would be diluted with water or chemically neutralized and pumped from the reprocessing facility to the injection facility or to interim storage in holding tanks.

Site. Deep well injection would require natural, intergranular fracture porosity or solution porosity formations, overlain by impermeable cap rock, such as shale. A minimum acceptable depth for disposal would be about 1,000 m (3,300 ft) (EPA 1973). The injection site must not conflict with either present or future resource development.

Synclinal basins would be particularly favorable sites for deep well liquid injection since they consist of relatively thick sequences of sedimentary rocks frequently containing saline ground water (Warner 1968). Ground-water movement within the injection formation would have to be limited, however, particularly vertical movement.

The lithological and geochemical properties of the isolation formation would have to be stable so that the behavior of the waste could be accurately predicted. In general, sandstone would be the most suitable rock type because it combines an acceptable porosity and permeability with chemically inert characteristics relative to the acid waste form.

The overall site area has not been determined yet, but would be greater than the 1270 ha (3140 acres) initial injection area and would depend on the maximum horizontal dimension of the injection area, the size of control zone required around the repository, and the total amount and type of waste to be injected.

Drilling System. The drilling rigs would be similar to those used in the gas and petroleum industries and would be portable for movement from one location to another on the site. Each complete rig would require a clear, relatively flat area, approximately 120 m x 120 m (400 ft x 400 ft) at each hole location (see Section 6.1.1).

Repository Facilities. The processing plant would be located on site as an integral part of the overall injection system. The basic repository facilities would be similar to those required for the very deep hole concept, as discussed in Section 6.1.1 (Bechtel 1979a).

Interim storage tanks similar to those described for the rock melt concept (Section 6.1.2) would be provided for surge capacity. The stainless steel tanks would have a combined capacity of about 10^6 liters (2.8×10^5 gal) which equals 3 months production. The tanks would be similar in design to those at the AGNS plant in Barnwell, South Carolina, which are contained in underground concrete vaults and provided with internal cooling coils and heat exchangers to prevent the waste from boiling.

An underground pipeway system would connect the reprocessing facility to the storage tanks and the injection facility. The pipe would be double cased and protected by a concrete shielding tunnel with leak detectors provided in the annulus of the pipe. The pipeway design would provide containment, monitoring, decontamination, maintenance, and decommissioning capabilities, primarily performed remotely. A heavy concrete and steel confinement building would provide containment for the well and injection operations and shielding for the radioactive systems.

Sealing Systems. The well hole would probably be sealed by a combination of borehole seals and backfilling, using a procedure similar to the one discussed for the very deep hole concept (Section 6.1.1).

Retrievability/Recovery. Liquid waste that had been injected might be partially retrievable by conventional well techniques. Although much of the waste might be physically or chemically sorbed by host geologic media, some species, in particular, 137 Cs, would be expected to remain in at least partially retrievable solution.

Shale Grout Injection

In the shale grout injection process, neutralized liquid waste or an irradiated fuel slurry would be mixed with a solids blend of cement, clay, and other additives, and the resulting grout would be injected into impermeable shale formations. The initial fracture in the shale would be generated by hydrofracturing with a small volume of water. The injection of waste grout into this initial fracture would generate sufficient pressure to propagate a thin horizontal crack in the shale. As injection of the grout continued, the crack would extend further to form a thin, approximately horizontal, grout sheet, several hundred feet across. A few hours after injection, the grout would set, thereby fixing the radioactive wastes in the shale formation. Subsequent injection would form sheets parallel to and a few feet above the first sheet.

The principal requirement for shale grout injection is that the hydrofracture, and hence the grout sheet, develops and propagates horizontally. Vertical or inclined hydrofractures could result in the waste gaining access to geologic strata near the surface, and even breaking out of grout at the bedrock surface itself. Theoretical analyses indicate that, in a homogeneous isotropic medium, the plane of hydrofracture develops perpendicularly to the minor principal stress (NAS 1966). Thus, a requirement for horizontal hydrofracturing is that the horizontal stresses exceed the vertical stresses.

On the basis of work at ORNL, approximately 40 injection wells would be required at each of five facilities. The activity level for the shale grout injection alternative is based on the reference concept (Schneider and Platt 1974) of 40 Ci/l activity in the initial grout. The acceptable gross thermal loading (GTL) could be assured by controlling the number of grout injections in the disposal formation. Depending on the fuel cycle, the maximum number of 2-mm (0.08-in.)-thick grout layers would be five to seven per injection site.

Site. A thick sequence of essentially flat-lying shale strata would be required for shale grout disposal, with in situ stress conditions favorable for the propagation of horizontal hydrofractures. Such conditions are generally found to a maximum depth of 500 to 1,000 m (1,650 to 3,300 ft). As with deep well liquid injection, the site would have to be located to preclude conflicts with resource development.

Shale deposits in the United States have been studied for suitability for underground waste emplacement (Merewether et al. 1973). The studies conclude that shale, mudstone, and claystone of marine origin in areas of little structural deformation, low seismic risk, and limited drilling are generally most promising. These include the Ohio shale of Devonian age in northern Ohio and the Devonian-Mississippian Ellsworth shale and the Mississippian-coldwater shale in Michigan. In the Rocky Mountain states, the Pierre shale and other thick shales of late Cretaceous age are also potential host rocks.

The overall site area for shale grout injection has not been determined yet, but it would be greater than the 1270 ha (3140 acres) initial injection area and would depend on the maximum horizontal dimension of the injection area and the size of the control zone required around the repository.

Drilling System. The drilling system for shale grout injection would be similar to that for deep well injection.

Repository Facilities. Repository facilities for shale grout injection would be identical to those for deep well injection with the exception of additional high-pressure pumps for fracturing and equipment related to mixing the grout with the liquid waste prior to injection (see Figure 6.1.19).

Sealing Systems. The repositories would be sealed in the same manner as deep well holes.

Retrievability/Recovery. Wastes disposed of by this concept would be essentially irretrievable because of the fast solidification and stability of the waste-grout mixture. Total recovery of the wastes would likely involve extremely difficult and extensive mining operations to excavate the rocklike waste form.

6.1.6.3 Status of Technical Development and R&D Needs

Present State of Development and Technological Issues

The basic techniques required for well injection of fluids and grouts have been developed in the course of many projects undertaken by the oil and chemical industries for the disposal of nonradioactive toxic and nontoxic wastes. In addition, limited disposal of radioactive waste grouts has been successfully completed at ORNL (ERDA 1977, Delaguna et al. 1968).

Geology. The geology of sedimentary basins in the United States has been examined extensively with a view to suitability for deep well liquid injection of radioactive wastes, and reports are available covering several areas.[a] In addition to these studies, a large

(a) See Repenning 1962, Sandberg 1962, Beikman 1962, Maclachlan 1964, Legrand 1962, Repenning 1959, Colton 1961, and DeWitt 1961.

volume of geologic data (stratigraphy, lithology, petrography) exists for potential disposal areas. These data have been gathered for basic geologic research or as a result of resource exploration and exploitation. However, the existing data are considered suitable for only conceptual, generic studies and identification of candidate sites.

Geohydrology. Modeling to predict waste extent and nuclide transport would be required for both liquid and grout injection. In the past decade, numerical modeling methods using finite-difference and finite-element techniques have been developed using available high-speed digital computers (Pinder and Gray 1977, Remson et al. 1971). Two- and three-dimensional fluid-flow techniques with thermal and stress dependency are available. Computer codes also exist for the analysis of radionuclide transport, including the effects of decay, adsorption, and dispersion (Burkholder 1976). However, these analytical techniques are limited because of an insufficient data base and incompletely defined constitutive parametric relationships.

State-of-the-art testing techniques include the use of multiple devices to isolate sections of the borehole. These devices provide for reduction in measurement error through improved control of bypass leakage. The multiple devices also help determine directional permeability (Maini et al. 1972). Multiple hole analyses are used to define the direction and magnitude and measure of rock mass permeability (Rocha and Franciss 1977, Lindstrom and Stille 1978). Because rock properties are directionally dependent, particular consideration must be given to methods of analyzing field data before a well injection site could be chosen.

Drilling and Injection Technology. The well injection disposal would require relatively simple engineering design, construction, and operation. Oil well drilling technology, fundamental to the concept, is available and well proven.

The deep well injection disposal method has been applied in the United States for natural wastes, in particular, oil-field brines, and for industrial wastes, such as steel pickle liquors, uranium mill wastes, and refinery and chemical process wastes[a]. The deepest waste injection well completed and operated to date was at Rocky Mountain Arsenal, where fractured Precambrian gneiss, at a depth of 3,660 m (12,000 ft), was used as the disposal formation (Pickett 1968).

Shale grout injections of remotely handled TRU wastes have been carried out at ORNL at a depth of about 275 m (900 ft) (ERDA 1977). Over 6.8×10^6 l (1.8×10^6 gal) of waste containing primarily ^{137}Cs (523,377 Ci) with a lesser amount of ^{90}Sr (36,766 Ci), together with minor quantities of other radionuclides have been injected over 10 years.

(a) Such applications are described in DeWitt 1961, Pinder and Gray 1977, Remson et al. 1971, Burkholder 1976, Maini et al. 1972, Rocha and Franciss 1977, Trevorrow et al. 1977, Lindstrom and Stille 1978, White 1965, Hult et al. n. d., Pickett 1968, Warner and Orcutt 1973, Lunn and Arlin 1962, Clebsch and Baltz 1967, Spitsyn et al. 1973, Capitant et al. 1967, and Roedder 1959.

Waste Preparation Technology. Liquid waste might require pretreatment to ensure compatibility with the rock. No operating injection facilities exist at present for high-level acid wastes. Pretreatment for most industrial wastes comprises filtration and limited chemical treatment. Since well injection is usually being pursued to reduce waste processing requirements, chemical treatment is minimal, and may include the addition of biocides and chloride to prevent plugging of the well from bacterial growth (Hartman 1968).

Waste preparation for shale-grout injection at ORNL has been the subject of extensive testing to develop an economical mix with good pumping and leach-rate characteristics (Moore et al. 1975, Hollister and Weimer 1968). Research indicates that the use of ash as a partial substitute for cement reduces costs and enhances strontium retention. Mixes incorporating various clays and grout shale have been tested. Leach rates of 3.2×10^{-5} g/cm^2/day for strontium and 2.1×10^{-6} g/cm^2/day for cesium have been obtained. The latter value is approximately equivalent to the leach rate for borosilicate glass (ERDA 1977).

Isolation and Safety. Isolation and safety analyses are based on

- Definition of source term (concentration, form, location, time)
- Characterization of pathway (transport velocity, chemical or physical changes, path length barriers, ecosystems involved)
- Exposure and "dose-to-man" calculations for both specific groups and total population.

A range of data values for the parameters can be analyzed to provide a probabilistic basis for the results. Methods involving modeling and analysis of failure processes have been employed for analyzing the performance of conventional disposal options (Logan and Berbano 1977) and would also be applicable to deep well injection concepts.

R&D Requirements

Since experience in the basic techniques required for well injection exists, the uncertainties associated with the design basis are related primarily to extrapolation of this experience to other waste forms, to other geologic settings, and to modified quantities and disposal rates. There are already techniques for preparing radioactive wastes in liquid or slurry form; however, there are uncertainties in formulating liquid wastes that would provide stability and compatibility with the disposal formation. For slurries, further R&D would be required for the development of optimum mixes, which would be related to the specific characteristics of the waste and disposal formation.

Geologic formations suitable for the injection of waste would have to be identified and verified on a site-specific basis. The exploratory techniques needed to do this are in an early stage of development, and would require further R&D with particular emphasis on verifying local geologic structure, establishing local and regional geohydrologic conditions, determining thermal and mechanical properties and in situ stresses, and locating and orienting discontinuities.

With the basic technology for injecting radioactive wastes into geologic strata already available, these research and development requirements can be categorized into several discrete areas of development, as described below.

System Data Base. It would be essential that the total R&D program be supported by a data base that covered all the components that could affect performance of the disposal system. The data base would cover the waste form, its modification, storage and injection, and the characteristics of the disposal formation from near to far field.

Development of Criteria for and Categorization of Siting Opportunities. The two types of well injection disposal methods, liquid and grout injection, would require significantly different but clearly definable disposal formation characteristics. Disposal site selection would have to proceed in stages, starting with the derivation and assembly of specific criteria, followed by successive narrowing of the field of choice to a specific site or sites. This approach would provide valuable generic hydrogeological data at an early stage for subsequent use in other R&D studies. The selection process could be undertaken initially using available geologic and hydrologic data and techniques. At the site-specific level, however, the use of yet-to-be developed "nonpenetrative" techniques might be required to minimize the amount of down-hole exploration.

Liquid and Slurry Wastes. A key facet of well injection is pretreatment of the liquid or slurry to a form that would be both compatible with the receiving formation and also the best use of the potential of that formation to fix and retain the nuclides. Optimum forms and requisite admixtures would have to be identified. The R&D program would have to proceed from the generic to the specific when the geochemistry of the disposal formation is known.

Techniques for Predicting the Configuration of Injected Wastes. Fundamental to the concept of "safe" disposal of waste is the necessity to predict, with a high degree of accuracy, the configuration that the injected wastes, whether liquid or grout-fixed slurry, would adopt in the disposal formation for both the short and long term. The technology should provide this capability.

For the liquid injection method, predictive capability is currently limited by the existing data base. Numerical simulation techniques are available, but these do not cover the range of conditions that might be encountered. Mathematical models for geohydrological and geochemical interaction studies would be needed.

"Nonpenetrative" Exploration Techniques. The presence of a drill hole could impair the isolation of a disposal site. At present, the majority of exploratory techniques require drilling at least one hole (and often several) to obtain reliable information from geological strata. R&D would be needed to develop nonpenetrative exploration techniques, similar to other geologic disposal methods.

Sealing Systems. It is assumed that the sealing system for well injection would have to meet the same time requirements for sealing penetrations that a mined repository must meet. The primary purpose of the seal is to inhibit water transport of radionuclides from the waste to shallow ground water or to the surface for an extended time period. Expansive concretes make the best seals under current technology and do so at an acceptable cost. However, current experience with seals, whether of cement, chemical, or of other materials, is only a few years old. Further development of sealing technology would, therefore, be required (Bechtel 1979a). For integrity to be maintained, the sealing material would have to meet the following requirements:

- Chemical composition - the material must not deteriorate with time or temperature when compared to host rock characterization.
- Strength and stress-strain properties - the seal must be compatible with the surrounding material, either rock or casing.
- Volumetric behavior - volume changes with changes in temperature must be compatible with those of enclosing medium.

The sealing system for well injection would consist not only of plugs within the casing, but also of material to bridge the gap between casing and competent rock not damaged by drilling. To minimize possible breaks in containment, rigorous quality assurance would be required during emplacement of several high quality seals at strategic locations within the borehole.

Research and development would be needed in two major areas - material development and emplacement methodology - to ensure complete isolation. Material development would include investigating plugging materials (including special cements), compatible casing materials, and drilling fluids. Because the seal would include the host rock, these investigations should include matching of plugging materials with the possible rock types. It is conceivable that different materials would be required at different levels in the same hole.

Emplacement methodology would have to be developed for the environment of the hole. Considerations would include operation in the aqueous environment, casing and/or drilling, and fluid removal. Because the emplacement methodology would depend on the type of material, initial studies of material development would have to precede emplacement methodology development. However, the two investigations would be closely related and would interface closely. In situ tests would have to be performed to evaluate plugging materials. Equipment developed would include quality control and quality assurance instrumentation.

Monitoring Techniques. In common with other methods of underground disposal, techniques would be required for monitoring the movement/migration of radioactive material from the point of emplacement.

Borehole Plugging Techniques. Borehole plugging techniques would require development at an early stage to permit safe exploration of candidate sites.

Implementation Time and Estimated R&D Costs

The R&D program described above is generic. Specific estimates for required implementation time and R&D costs would depend on the details of the actual development plan, and are deferred pending plan definition.

Summary

Major uncertainties, shortcomings, and advantages of the concept are summarized below:

- The concept is not compatible with the multi-barrier philosophy, relying only on a potentially non-inert waste form and the geology.
- Performance assessment and siting technology for HLW injection are essentially non-existent.
- Retrievability, technical conservatism, and adequate design margins do not appear possible due to the diffuse nature of the emplaced material.
- The emplacement technology is considered to be essentially available.

6.1.6.4 Impacts of Construction and Operation (Preemplacement)

In some respects the environmental impacts of the well injection concepts are better understood than the impacts from the other disposal alternatives. This is because of their current use--deep well by the oil and gas industry to dispose of chemical waste and shale grout injection by the Oak Ridge National Laboratory to dispose of remotely handled TRU wastes. Potential use of well injection for disposing of long-lived or high-level radioactive waste, however, has not been demonstrated.

Although quantitative estimates of environmental impacts of well injection have not been made, it is expected that many of the impacts would be essentially the same for the two reference concepts.

Health Impacts

Radiological Impacts. The radiological impacts from routine operations during most phases of well injection disposal (e.g., reactor spent fuel storage, and intermediate spent fuel storage) are expected to be the same as those for a mined geologic repository. However, the extra operation to reprocess spent fuel from the once-through fuel cycle to produce a liquid solution or grout could be expected to add to the radiological impacts. Quantitative estimates of these impacts are not available at this time. Likewise, the radiological impacts associated with the transportation of wastes are expected to be similar to those for a mined geologic repository, with the exception of transporting HLW from the reprocessing plant. Since, for the reference repositories, the injection facility is adjacent to the reprocessing plant, the need to transport HLW is eliminated, which thereby reduces the corresponding radiological impact.

Unavoidable environmental effects of the well injection option would include operational radiation doses to facility workers involved in injection or maintenance and repair. Design and operational procedures would be directed to reducing doses to the lowest levels possible. At the ORNL remotely handled TRU waste facility the radiation exposure per man per grout injection has averaged 0.025 rem during injection operations and 0.188 rem during preinjection maintenance (ERDA 1977). However, the data are not sufficient to determine whether these occupational exposures would be applicable to an HLW repository. Accident scenarios may be conveniently divided into surface and subsurface events. Surface operating accidents would include pipe ruptures and spills, failure of transfer or injection pumps, and loss of necessary cooling to the storage tanks. To minimize risk, normal nuclear engineering design strategies would be required, with redundancies incorporated into all critical systems and components (for example, pumps, power supply, and monitoring equipment). Subsurface accidents, for which contingency plans would have to be prepared, would include well-pipe rupture, equipment failures, uncontrolled fracture development (shale grout injection), and penetration of waste through the containment formation due to highly permeable features, abandoned or poorly sealed wells, or exploration or monitoring of drill holes. Site exploration and analyses would be directed toward minimizing the probability and the effects of subsurface failures.

Presently, there are no quantitative estimates of the radiological impacts of such accidents to occupational personnel, nonoccupational personnel, or the ecosystem. Furthermore, since the waste would be in a nonsolid form for well injection, the radiological impacts are not expected to be similar to those resulting from accidents at a mined geologic repository.

Nonradiological Impacts. Little formal study has been completed on the nonradiological health effects of the well injection disposal process. In general, predisposal activities, such as fuel handling, storage, transportation, and reprocessing, for both reference concepts would be the same as for a mined geologic repository. Pretreatment of the disposal formation with acid, however, might be required. Although potential impacts have not been quantitatively assessed, it can be concluded that nonradiological health effects would result from handling this hazardous material.

Because wastes injected into the wells would have to be in liquid or grout form, two important differences are anticipated between well injection and mined geologic disposal. First, the well injection disposal site would have to be at the same place as the reprocessing facility. Colocating these facilities would minimize the transportation requirements and associated risks. It would also reduce some of the nonradiological impacts associated with transportation activities.

Second, well injection would involve surface and subterranean activities with different hazards than those associated with mined geologic disposal--formation drilling and fracturing, compared to large-scale excavation, are the principal below-ground activities that could lead to nonradiological health impacts. Preparing the wastes for disposal would involve facilities designed to mix the wastes with clay, cement, and other additives for the shale-grout method. For the liquid injection process, more limited mixing facilities would be needed. In either case, studies completed to date have not identified significant nonradiological impacts for these activities under routine operating conditions. Under abnormal conditions, pipe ruptures and spills, failure of injection pumps, and other problems discussed under radiological impacts could lead to nonradiological impacts as well.

Natural System Impacts

Effects on the ecosystem near a well injection disposal site would be similar to those associated with any heavy engineering project. In considering these impacts, it must be remembered, however, that the disposal site would include reprocessing and disposal facilities.

Ecological impacts from these processes are categorized into preconstruction and post-construction activities. Initial construction activities would involve clearing vegetation, drilling, and geophysical surveying. Impacts of these initial activities would affect vegetation, soil, water, and other resources to varying degrees depending on the characteristics of the specific site being developed. Impacts of this type of activity are evaluated for specific sites.

Construction impacts would include those of a reprocessing facility, as described in Chapter 4. Construction of facilities to prepare the wastes for injection, as described above, would also be needed.

Postconstruction, or operational, nonradiological ecological impacts would be more limited than those of preconstruction and construction activities. Many operational activities would occur below the surface. Ecological impacts from these activities could occur if some of the fluids injected into the well were to enter the ground-water system and were transported to the biosphere or otherwise affected aquatic resources. Surface runoff or material spilled on the surface could also cause localized ecological impacts.

Socioeconomic Impacts

Socioeconomic effects from constructing and operating a well injection repository would be felt most intensely in the immediate vicinity of the facility. In general, impacts would be representative of those of a major engineering facility. No quantitative data exist on the construction or operational employment requirements of a well injection disposal system. Impacts, however, should be similar to those described for the very deep hole concept (see Section 6.1.1.6). In addition, socioeconomic impacts associated with the reprocessing facility would be felt at the disposal site. These impacts are discussed in Section 4.7. In analyzing these discussions, it must be remembered that colocation would lead to a greater concentration of impacts at the disposal site, but at the same time would reduce the number of separate nuclear facilities constructed.

Aesthetic Impacts

Aesthetic impacts for the well injection disposal option would be similar to those of other subsurface disposal methods except for the presence of the reprocessing facility at the disposal site. Again, colocating facilities could increase the impacts at the chosen site, but the fact that only one site is needed suggests an overall reduction in aesthetic impacts.

Aesthetic impacts could be accurately assessed only within the context of a specific site. In a general context, however, aesthetic impacts related to drilling and other geologic activities are covered in the aesthetic impact discussions for mined geologic repositories (Section 5.5) and the very deep hole concept (Section 6.1.1.6). Aesthetic impacts of reprocessing facilities are discussed in Section 4.7.

Resource Consumption

Suitable well injection sites would be sedimentary basins, which are frequently prime areas for fossil fuels. However, after the wastes had been safely emplaced, geologic exploratory activities in the vicinity of the site would have to be restricted. It has been suggested that potentially usable minerals from the zone of influence of the repository would be inventoried before implementation would begin. On the other hand, the disposal zone itself could be considered a resource for which alternative uses might be found, for example, storage of freshwater or natural gas.

Other resources consumed in the well injection process would include energy for transportation, processing, and disposal. Land would be required for the reprocessing and disposal facilities. For the shale-grout disposal method, clay, cement, and other materials would be needed. No critical material, other than fuel, would be consumed by well injection disposal.

International and Domestic Legal and Institutional Considerations

Implementation of the well injection option would require two important policy decisions that could be shaped by institutional forces. First, the process does not lend itself to handling spent fuel from reactors. Processing would be needed to transform this material into a form that could be readily injected into the well. The reprocessing approach most often proposed contravenes the current U.S. position against reprocessing. This would have to be resolved before well injection disposal could be implemented.

The second policy decision stems from the need to locate the disposal facility and the fuel reprocessing plant at the same site. Although such a system would be effective in limiting liquid waste transportation, it is likely that neither facility would be optimally located. It would have to be decided whether the benefits of well injection disposal outweigh potential disadvantages of such colocation. Obviously, such a decision would have to be made in light of domestic institutional considerations.

Another aspect of the well injection concept that could foster concern is the need to obtain records of previous drilling activities. States typically maintain such records and generally oversee drilling programs. If this disposal option were implemented, information would be needed and procedures would have to be established to evaluate data from adjacent well sites. The relationship between existing regulatory activities and the well injection disposal process would have to be defined prior to implementation.

Aside from the issues outlined above, the legal and institutional considerations of this option would be similar to those of the mined geologic repository discussed in Section 5.5.

6.1.6.5 Potential Impacts Over Long Term (Postemplacement)

An unavoidable long-term impact of well injection waste disposal is that alternative storage or disposal applications for the site are eliminated. Examples of possible uses are natural gas storage, freshwater storage, and disposal of other wastes of lower or shorter-lived toxicity. In addition, as noted earlier, exploration for natural resources and subsequent mining in a large area around the disposal facility would be subject to control. The extent of exclusion and limited activity buffer zones would depend on the characteristics of the disposal formation, and in particular, its hydrologic and geochemical conditions. Finally, evidence exists that injection of wastes into certain formations could potentially lead to seismic activity and earthquakes.

Potential Events

Natural Events. The long-term leaching and transportation of radionuclides in the ground-water system to the biosphere would be a fundamental pathway in the well injection concept, as it is with all geologic concepts. Assessment of the environmental impact would require predictive modeling of the rock mechanics, hydrology, and geochemistry of the disposal and containment formations, together with an adequate data base to characterize the biosphere. The disposal area would be selected to minimize the risks from seismic and volcanic activities and their effect on the hydrologic regime. Seismic events could induce tectonic effects within the disposal area, causing permeability and flow changes. Volcanic activity could result in catastrophic breach of the containment formation, or could generate unacceptable, thermally induced flow patterns. The risk of meteorite impact would be similar to that for a mined geologic repository; however, with deep-well liquid disposal, the waste would be in a more mobile form. The impact of gross changes, such as climate variations or polar ice melting, would, in general, depend on their effect on the hydrologic regime. Increased erosion (because of glaciation, for example) could reduce the cover of the disposal formation.

An impact of potentially major significance is the increased chance of an earthquake that could result from injecting waste material into rock formations. A relationship between deep well liquid injection and increased seismicity has been suggested (Evans 1966) in connection with earthquakes at Denver and injection at the Rocky Mountain Arsenal well. Other studies (Hollister and Weimer 1968, Dieterich et al. 1972) have shown that deep well injections in the Rocky Mountain Arsenal Range have been instrumental in producing seismic events. Obviously, such concerns are significant and would have to be seriously evaluated for specific sites. Knowledge of the in situ stress state for both concepts would be needed before proceeding with the well injection option because of the chance of earthquakes developing. The depth of shale grout injection would be limited by the requirement that vertical stresses be less than horizontal stresses.

Manmade Events. Exclusion and controlled-use buffer zones would be set up around an injection facility. Nevertheless, the risks associated with drilling into a waste-liquid or grout disposal formation would have to be considered. Changes in the surface and subsurface hydrologic regime of the area, because of reservoir construction, deep excavation and construction, and resource exploitation outside the buffer zone, would require analysis.

The geologic formation in which a well injection repository would be located would have to be bounded by impermeable strata and free of water-transmitting faults. Such formations occur in the sedimentary basins in the U.S., and it is these basins that oil and gas companies are exploring for petroleum and natural gas. This exploration could cause a major safety problem by connecting waste disposal zones with aquifers.

Potential Impacts

As with the mined geologic repository, the principal pathway for release of radionuclides to the biosphere in the long term would be by ground-water transport. It is believed, however, that the likelihood of ground water reaching the injected waste is extremely small.

The only quantitative estimates on the movement of radionuclides via ground water transport are from ORNL's experience with grout injection of remotely handled TRU waste into shale (ERDA 1977).

The maximum quantity of activity that could be leached from a single grout sheet was calculated, using data presently available (ERDA 1977). This sheet would have a volume of about 28,300 m^3 (1 million ft^3) and could contain as much as 500,000 Ci of ^{90}Sr (if a maximum waste concentration of 5 Ci/gal is assumed) and an equal amount of ^{137}Cs. Leach data indicate that the 6-month leach rate of radionuclides from cured grouts would not exceed 6.2×10^{-5} Ci/month of ^{137}Cs per sq ft of leached area, 1.7×10^{-3} Ci/month-ft^2 of ^{90}Sr, 5.5×10^{-7} Ci/month-ft^2 of ^{244}Cm, and 5.6×10^{-10} Ci/month-ft^2 of ^{239}Pu. If the entire grout sheet surface were exposed to water flow, a maximum of 62 Ci/ month of ^{137}Cs, 1700 Ci/month of ^{90}Sr, 0.6 Ci/ month of ^{244}Cm, and 6×10^{-4} Ci/ month of ^{239}Pu would be leached. If the water flow is assumed to be 0.5 ft/day, the calculated concentration of ^{239}Pu in the water would be approximately 1×10^{-6} Ci/ml (less than the concentration guide for this isotope in uncontrolled areas). The shale surrounding the grout sheets has considerable ion-exchange capacity for cesium and strontium; a calculation yields rate of movement of leached cesium and strontium through the shale that would be so low that these nuclides would be transmuted by radioactive decay long before they approached the surface. The small quantity of ^{244}Cm that might be leached would also be retained by the shale.

6.1.6.6 Cost Analysis

Capital, operating, and decommissioning costs of well injection disposal have not been estimated. However, since well injection disposal would not require costly mining operations, it could offer a low-cost means of disposal compared to mined repositories.

Cost data are available from ORNL (ERDA 1977) for a site-specific application of grout injection disposal of RH-TRU. Estimated capital costs for a new waste shale fracturing disposal facility, adjusted to 1978 dollars, are $6.0 million. Annual operating costs are estimated at $110,000. No data are given for decommissioning costs. The costs are estimated for a facility to perform removal of large volumes of mobile radioactive wastes from existing near-surface storage facilities at Oak Ridge.

6.1.6.7 Safeguard Requirements

Because of the restrictions concerning the transportation of high-level liquid waste, which require the injection facility to be colocated with the fuel reprocessing plant, the accessibility to sensitive materials would be extremely limited. However, this waste disposal system would probably be used in a uranium-plutonium recycle fuel cycle so there would be incremental increases in accessibility in other parts of the fuel cycle similar to most recycle scenarios. In addition, the difficulty of retrieving material once it had been successfully disposed of would increase the difficulty of diversion and the waste form (liquid) would complicate the transportation and handling problems for a potential diverter. The deep well injection repository would require additional safeguards since at least partial retrieval by drilling and pumping might be possible. Material accountability would be enhanced by ease of sampling and measurement of liquids, but gross accountability (i.e., gallons vs canisters) would be slightly more difficult than for the reference mined geologic concept.

See Section 4.10 for additional discussion of predisposal operations safeguard requirements.

6.1.7 Transmutation

6.1.7.1 Concept Summary

The primary goal of waste disposal has been stated as protection of the public. This would be achieved in mined geologic disposal by containing the high-level radioactive waste for the time period during which it retains significant quantities of potentially harmful radionuclides. One alternative to this approach is to selectively eliminate the long-lived radionuclides by converting or transmuting them to stable or short-lived isotopes. This approach would shorten the required containment period for the remaining waste. Shortening the containment period would increase confidence in predicting the behavior of the geologic media and reduce the requirements on the isolation mechanism. Thus, an attractive feature of transmutation is that it has the potential to reduce the long-term risk to the public posed by long-lived radionuclides.

In the reference transmutation concept, spent fuel is reprocessed to recover the uranium and plutonium. The remaining high-level waste stream is partitioned into an actinide stream and a fission product stream. The fission product stream is concentrated, solidified, vitrified, and sent to a terrestrial repository for disposal. In addition, actinides are partitioned from the TRU-contaminated process waste streams from both the fuel reprocessing plant and the mixed oxides fuel fabrication plant. The waste actinide stream is combined with recycled uranium and plutonium, fabricated into fuel rods, and reinserted into the reactor. For each full power reactor year, about 5 to 7 percent of the recycled waste actinides are transmuted (fissioned) to stable or short-lived isotopes. These short-lived isotopes are separated out during the next recycle step for disposal in the repository. Numerous recycles result in nearly complete transmutation of the waste actinides.

A disposal system that uses transmutation would have the environmental and health impacts associated with the recycle of uranium and plutonium and with the partitioning of the actinides from the waste stream. If uranium ahd plutonium recycle were adopted for other reasons transmutation would be more feasible but would still involve additional impacts. For example, highly radioactive fuel elements containing recycled waste actinides would need to be fabricated, handled, and transported. The additional facilities and waste treatment processing steps required could be expected to increase effluent releases to the environment, the occupational exposure, the risk of accidents, and costs. Since only about 5 to 7 percent of the recycled waste actinides would be transmitted to stable isotopes in each reactor irradiation, numerous recycles would be required with attendant additional waste streams.

6.1.7.2 System and Facility Description

System Options

The reference concept was selected from several available options. These options are listed in Figure 6.1.20 for each major step in a flowsheet using transmutation.

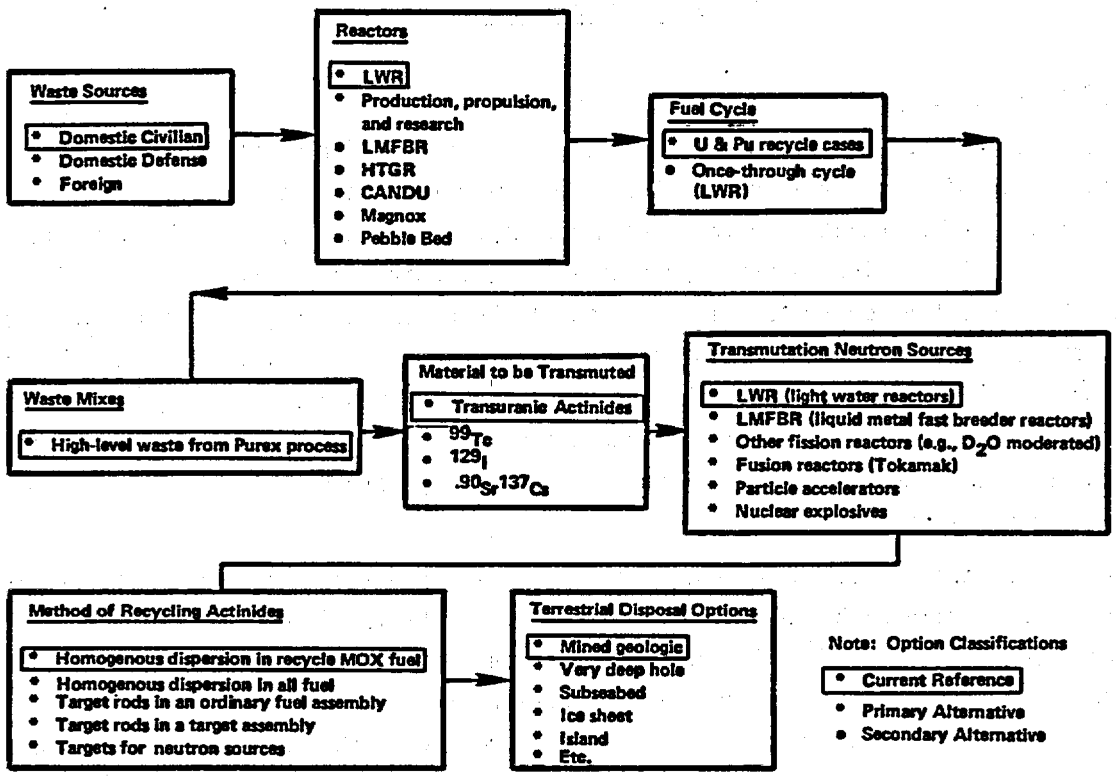


FIGURE 6.1.20. Major Options for a Waste Disposal Alternative Using Transmutation

The reference concept was selected somewhat arbitrarily to be used as a basis for comparison and to help identify the impacts associated with a typical transmutation fuel cycle. If transmutation were selected as a candidate alternative for further research and development, considerable study would be required to optimize the available alternatives. Additional information concerning the advantages and disadvantages of the many process options is available in sources listed in Appendix M.

Waste-Type Compatibility

Transmutation would be applicable to only those fuel cycles that involve the processing of irradiated nuclear fuel, e.g., the recycle of uranium and plutonium. In that context, transmutation would not apply to once-through fuel cycles. It could be used with both commercial and defense waste, although little work has been done concerning defense wastes.

Waste-System Description

The fuel cycle and process flow for the reference concept are shown in Figure 6.1.21. The cycle begins with the insertion of a reload of fuel into the reactor. The reload is two-thirds fresh enriched UO_2 and one-third recycle mixed oxide (MOX) fuel, which has all the waste actinides (i.e., neptunium and other transplutonics) homogeneously dispersed in it.

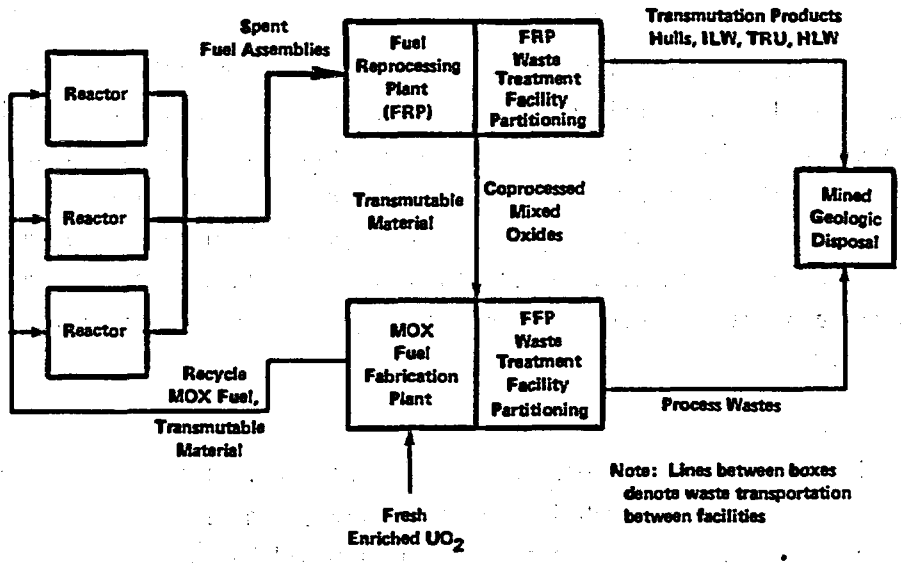


FIGURE 6.1.21. Partitioning-Transmutation Fuel Cycle Diagram

The cycle continues by:

- Irradiating the reload to a burnup of 33,000 MWd/MTHM
- Discharging and decaying the reload for 1-1/2 years
- Reprocessing the UO_2 and MOX fuels together
- Sending the TRU-contaminated wastes to the fuel reprocessing plant waste treatment facility (FRP-WTF) for partitioning
- Returning the recovered TRU and the TRU-depleted wastes to the reprocessing plant
- Combining the recovered actinides with the processed MOX and transporting the mixture to the refabrication plant, after a 6-month delay
- Adding sufficient uranium to the MOX product to achieve the desired end-of-cycle reactivity. (This product is in powder form and contains the waste actinides.)
- Refabricating the MOX product
- Sending the TRU-contaminated wastes from refabrication to the fuel fabrication plant waste treatment facility (FFP-WTF) for partitioning
- Returning the stream of recovered actinides to the fabrication plant
- Incorporating the recovered actinides with MOX recycle streams within the facility
- Sending TRU-depleted wastes to a mined geologic repository.

Simultaneously, the fresh enriched UO_2 fuel is fabricated in a separate facility. At this point, the cycle is completed with the fabricated fuels being inserted into the reactor. The details of the waste treatment facility (WTF) process and plant design are given in Tedder et al. (1980) and Smith and Davis (1980).

Predisposal Treatment

In a fuel cycle involving transmutation, it would be necessary to partition the materials to be recycled and transmuted. The partitioning flowsheet would have two fundamental steps. The first would be to separate the actinides from other materials and the second would be to recover the actinides in a relatively pure form. Actinides would be separated by various methods and would originate from many sources, including high-level waste, dissolver solids, cladding, filters, incinerator ashes, salt wastes, and solvent cleanup wastes. The extractable actinides from these operations would be sent to actinide recovery, where they would be partitioned and purified.

Facilities Description

There are four facilities in the reference fuel cycle that process the actinides: the fuel reprocessing plant (FRP), the fuel fabrication plant (FFP), and a colocated waste treatment facility (WTF) for each. The purpose of the two WTF's would be to recover a high percentage of the actinides that would ordinarily be delegated to process wastes.

The FRP-WTF and FFP-WTF would have the following common process capabilities:

(1) Actinide recovery

(2) Cation exchange chromatography (CEC)

(3) Acid and water recycle

(4) Salt waste treatment

(5) Solid alpha waste treatment.

In addition, the FRP-WTF would have high-level liquid waste and dissolver solid waste treatment process capabilities. The WTF facilities would be constructed on sites about 460 m (1,500 ft) from the FRP and FFP, but still within a fuel cycle center that would allow common services and utilities for the entire center. Additional detailed design and cost information is available in Smith and Davis (1980).

Since transmutation would take place in the reactor itself, no special facilities would be required, although the irradiation levels of the recycle fuel require that the fuel assemblies be handled remotely. Because transmutation would eliminate only a specific segment of the waste, all the facilities required for conventional terrestrial disposal, e.g., a mine geologic repository as described in Chapter 5, would also be necessary in this fuel cycle. The use of transmutation would not significantly change the total amount of waste or the necessary throughput of waste disposal facilities.

Retrievability/Recovery

The segment of waste disposed of in the mined geologic repository would exhibit the same characteristics discussed in Chapter 5 of this report.

6.1.7.3 Status of Technical Development and R&D Needs

Only the referenced use of transmutation - recycling, using commercial nuclear reactor fuels, to minimize the actinides contained in radioactive waste - is discussed here. Part of the R&D associated with transmutation would be the continued investigation of other useful applications of the process. There are several other waste constituents that could be transmuted.

Present Status of Development

Transmutation represents an advanced processing concept that would require R&D work before incorporation into any system. There are still uncertainties associated with many of the subsystem details. Although the concept is technically feasible, it should be recognized that the required design bases have not been sufficiently refined to permit construction of full-scale facilities. For some partition subsystems, laboratory experiments have been developed to demonstrate technical feasibility only. Only preliminary material balance calculations have been performed and, in most cases, no energy balances are available.

A number of transmutation devices for converting various nuclides to other more desirable forms have been studied. Neutron irradiation can be carried out with nuclear explosive devices, fission reactors, or fusion reactors. Accelerators can provide charged particle beams of protons or heavier ions for producing neutrons for irradiating selected nuclides. For the actinides, the most practical transmutation occurs by irradiation by a fission reactor neutron source. The estimated actinide transmutation rate utilizing commercial light water reactors is about 6 percent for each full-power year that the actinides are in the reactor (EPA/MITRE 1979).

There are four principal methods for recycling actinides in light water reactors: (1) dispersing the actinides homogeneously throughout the entire fuel reload, (2) dispersing the actinides homogeneously in only the mixed-oxide fuel, (3) concentrating the recycled waste actinides in target rods within an otherwise ordinary fuel assembly, and (4) concentrating the recycled waste actinides in target rods that are then used to make up a target assembly. In the first two methods, the actinides include all of the plutonium generated in the reactor. In the second two methods, plutonium (an actinide) is excluded from the targets but is recycled in a mixed-oxide fuel. On the basis of preliminary qualitative evaluation, it would appear that the second recycle mode, homogeneous dispersal of the actinides in the mixed-oxide fuel, is preferred over the others (Wachter and Croff 1980).

Technological Issues

The effect of a transmutation recycle, as opposed to the uranium and plutonium recycle mode, on the various elements of a conventional fuel cycle depends largely on two factors--the transmutation rate in the reactors and the manner in which the transmutation reactors are decommissioned as the cycle is eventually terminated. Important technological issues are:

- The use of commercial power reactors as transmutation devices might result in fissile penalties, reactor peaking problems, reduced reactor availability, and increased operating costs.
- Because of increased concentrations of radioisotopes with high specific activities, and/or modifications of existing systems due to changes in requirements, transmutation recycles could require additional containment systems to limit the release of radioactivity at the reactor site to acceptable levels.
- Many transmutation cycles would increase fuel handling requirements because of the more frequent insertion and removal of fuel and transmutation targets from the reactor core. Most transmutation cycles would result in increased shielding requirements both for fresh and spent fuels and transmutation targets.
- Decommissioning and disposal of reagents from partitioning and transmutation facilities would be complicated by the increased demands for shielding, multiple chemical processes, and waste streams.

The duration of the transmutation cycle is important in estimating its overall effectiveness in reducing the total radiotoxicity of transmutable elements in the environment. Premature termination of the transmutation cycle could actually increase the radiotoxicity of the wastes. This is because the resulting inventory sent to a final disposal system might have more activity than it would if the transmutation cycle had not been initiated.

R&D Requirements

The R&D requirements for partitioning would involve specific near-term subtasks to clarify points of uncertainty in the current process parameters and techniques. However, to fully develop and demonstrate actinide partitioning, a program would have to include additional process research and development, a cold (nonradioactive) testing facility, equipment development and testing, and pilot plant design, licensing, construction, testing, and operation.

Transmutation R&D would include specific nuclide cross section measurements, reactor physics calculations, and irradiation to full burnup of test fuel assemblies to verify calculations. The irradiation tests would also serve to confirm the design and fabrication of the fuel assemblies and their compatibility with and performance in the reactor during power operation.

The design, construction, and testing of a prototype shipping cask made from the relatively unconventional materials proposed might also be required. Specific aspects of cask technology that might require attention are: techniques for industrial fabrication of special shielding materials, such as B_4C/Cu and LiH, investigation of the ability of the cask using such materials to conduct the heat from the fuel contents, and the effect of the unusual construction materials on safety considerations in cask design.

Finally, continuing overall studies to define the preferred methods of operating the fuel cycle and the impacts and benefits of this operation would be of primary importance.

Implementation Time

The long lead time for implementing this alternative is based on the orderly development of a commercial scale partitioning plant, which would be expected to take about 20 years. The first 10 years would be devoted to partitioning research and the development and testing of a pilot plant, as reflected in Table 6.1.20. All of the R&D programs involving transmutation, fuel assembly and shipping cask development, and system studies could be accomplished in concurrence with the partitioning schedule.

Estimated R&D Costs

Table 6.1.20 identifies estimated R&D costs necessary to demonstrate the transmutation of actinides. It does not include costs associated with providing a commercial scale partitioning plant, the necessary modifications to the fuel fabrication facility and light water reactors, or a transportation system required to utilize the partitioning-transmutation of actinides as a waste disposal alternative.

Summary

Major uncertainties, shortcomings, and advantages of the concept are summarized below:

- The concept is actually a method of waste treatment or conversion to a more benign form; it is not an independent disposal method.
- Additional waste streams during the process are generated so that the actual volume of waste for isolation is greater than without it.
- The technology for efficient transmutation (waste partitioning and advanced reactors) are considered to be long-term achievements.

TABLE 6.1.20. Estimated Transmutation R&D Costs And Implementation Time

	Cost, $ million	Time Span, years
Partition R&D (Includes Pilot Plant)	560	10
Transmutation R&D	16	15
Fuel R&D	80	15
Transportion	56	10
System Studies	8	Continuous

6.1.7.4 Impacts of Construction and Operation (Preemplacement)

As described in Section 6.1.7.1, the transmutation option would include elimination of certain long-lived radioactive wastes and the disposal of the remaining waste material in a mined geologic repository. The potential benefits of transmutation that would be realized for the lower levels of long-lived hazardous material are discussed in Section 6.1.7.5, while short-term impacts of construction and operation are discussed here. Because these short-term impacts include those of a mined geologic repository, impacts identified in Section 5.6 must be considered a part of this option. In addition, impacts associated with reprocessing and discussed in Section 4.7 would occur.

Because transmutation is a waste processing option involving extra waste treatment steps, a meaningful impact analysis is possible only when a transmutation system is compared with a reference processing and disposal system. In the following analysis, the reference system includes waste reprocessing and final disposal in a mined geologic repository.

Another important factor in this discussion is that impacts attributed to one plant generally relate to a reprocessing plant handling 2000 MTHM per year and a fuel fabrication plant handling 660 MTHM per year. Such a hypothetical plant provides the basis of much of the information used in this analysis (Blomeke et al. 1980, Fullwood and Jackson 1980, Logan et al. 1980). Depending on the actual amount of nuclear wastes generated, several of these plants could be constructed.

Health Impacts

Radiological Impacts. The increased frequency of waste handling and transportation activities associated with the transmutation option suggests that it would result in increased radiation exposures compared with the mined geologic repository option.

ORNL estimated the radiological occupational impact of the reference concept based on routine exposure, maintenance exposure, and anticipated abnormal occurrences (Fullwood and Jackson 1980). Table 6.1.21 presents the collective dose rates calculated for the four facilities included in the study. The values range from a low of 3 man-rem/plant-year for an abnormal occurrence in the FFP-WTF to a high of 230 man-rem/plant-year for routine and maintenance exposure in the FFP.

The radiological exposure to the general public arising from routine operations is a consequence of the fact that the facilities would have to provide fresh air for the workers and vent gases to the atmosphere. In spite of elaborate air-cleaning practices and equipment, small amounts of radioactive materials would be discharged into the atmosphere; the amount varying with the chemical species. Estimates have been made for the amounts of radioactive materials that are expected to be discharged from each plant (Fullwood and Jackson 1980). The resulting exposures, based on these estimates, are presented in Table 6.1.22. The values range from 680 to 736 man-rem/plant-year for the Reference Facility and the P-T respectively.

TABLE 6.1.21. Annual Routine Radiological Occupational Dose

Facility		Exposure, man-rem/plant-year		
		Routine	Operation Maintenance	Abnormal
FRP	(1)	220	220	10
FRF-WTF	(2)	220	220	10
FFP	(3)	230	230	10
FFP-WTF	(4)	90	90	3
Reference Facility	(1) and (3)			
P-T	(1-4)			

The more significant of the postulated accidents have been analyzed as to the resulting effects on the plant workers. In general, individual worker exposure would exceed public exposure because of closeness to the accident. Isotopic differences between the two cycles would result in small differences in exposure, so there is negligible distinction between the Reference and the P-T cycle, except that the Reference Facility does not contain the two WTF's. The totals for the component facilities are presented in Table 6.1.23. The details of the accidents and other assumptions are given in Fullwood and Jackson (1980).

Table 6.1.24 presents corresponding data for the non-occupational consequences of the postulated accidents.

TABLE 6.1.22. Annual Routine Non-Occupational Dose

Process Stage	Exposure, man-rem/plant year	
	Ref. Facility	P-T
FRP	680	730
FRP-WTF	-	5.3
FFP	7×10^{-3}	1.7×10^{-2}
FFP-WTF	-	0.55
Totals	680	736

TABLE 6.1.23. Occupational Radiological Exposure--Abnormal

Conditions Facility	Exposure, man-rem/plant year
FRP	1.3×10^{-2}
FRP-WTF	1.3×10^{-2}
FFP	4×10^{-2}
FFP-WTF	7×10^{-3}

Besides the plants and processes another major activity in the fuel cycle would be transportation links for fresh fuel movement, spent fuel movement, powder movement between the FRP and FFP, and waste movement from the FRP-FFP complex to the repository and disposal area. Table 6.1.25 presents data resulting from accident analyses of the six transportation steps considered for the two fuel cycles.

Nonradiological Impacts. Nonradiological impacts would result from two factors that are unique to the transmutation alternative. First, the partitioning process would require additional facilities at the reprocessing plant and at the MOX fuel fabrication facility. Second, the nature of the wastes that would be generated by transmutation dictates increased transportation activities.

TABLE 6.1.24. Non-Occupational Radiological Exposures--Abnormal

	Exposure, man-rem/plant year	
Process Stage	Ref. Facility	P-T
FRP	5×10^{-3}	5×10^{-3}
FRP-WTF	-	6×10^{-5}
FFP	3×10^{-5}	3×10^{-5}
FFP-WTF	-	6×10^{-5}
Reference Facility	5×10^{-3}	
P-T		5.2×10^{-3}

TABLE 6.1.25. Transportation Non-Occupational Radiological Exposures--Abnormal

Transportation Step	Exposure, man-rem/plant year	
	Ref. Facility	P-T
Spent Fuel	2.3×10^{-3}	3×10^{-3}
Powder	2.3×10^{-10}	3×10^{-10}
Fresh Fuel	6×10^{-5}	3×10^{-5}
Cladding Hulls	1.2×10^{-2}	1.3×10^{-2}
HLW	8×10^{-4}	6×10^{-4}
NM-HLW	1×10^{-1}	9.8×10^{-2}
Totals	1.1×10^{-1}	1.1×10^{-1}

A closer examination of the first factor reveals that the additional partitioning facilities would be colocated at reprocessing and fuel fabrication sites. These incremental changes are analyzed as they would affect operational, environmental, and resource considerations.

Regarding the second factor, transportation impacts, the relatively small carrying capacity of the canisters that would be used to transport the fresh and spent fuel means more trips per unit of fuel than with options involving unpartitioned wastes. Furthermore, more waste would be generated. This would lead to more transportation impacts. It is estimated that the facilities included in this option would process 2,000 MTHM per plant per year. This means an estimated nine trips involving hazardous material would have to be made each day, as compared with an estimated seven trips per day for fuel reprocessing without transmutation (Fullwood and Jackson 1980). Although the increased emissions, chance of derailment, and community concern associated with more intensive transportation could not be accurately determined until a specific disposal system is proposed, it is recognized that transportation impacts would be greater than those for the reprocessing-only case.

Nonradiological health effects would occur as a result of construction and operation activities. In spite of scrubbers and other air-cleaning devices, small amounts of hazardous materials would be discharged into the atmosphere. There would be two main sources of these pollutants: the chemical processes themselves and the auxiliary services, primarily the steam supply system, which is assumed to burn fuel oil. Table 6.1.26 presents the annual health effects for transmutation. The data are based on estimates for the Allied General Nuclear services plant at Barnwell, South Carolina, but are scaled to allow for the larger size of the transmutation facilities. The health effects were estimated from epidemiological studies on SO_2 and its relationship to the other pollutants.

The increased transportation required for the transmutation alternative suggests a greater likelihood of occupational and nonoccupational hazards than with options not involving partitioning. Unlike radiological impacts, nonradiological concerns should not vary significantly from those of an industrial facility not involved in nuclear activity.

TABLE 6.1.26. Summary Effects (Per Plant-Year) of Non-Radiological Effluents (Fullwood and Jackson 1980)

Plant	Premature Deaths/yr Reference Facility	Premature Deaths/yr Transmutation	Permanent Disabilities/yr[a] Reference Facility	Permanent Disabilities/yr[a] Transmutation
FRP	4	4	14	14
FRP-WTF	--	7	--	21
FFP	0.2	0.2	0.6	0.6
FFP-WTF	--	3	--	9
Totals	4.2	14.2	14.6	44.6

(a) Based on disabilities lasting longer than 6000 person-days.

Probably the single most important nonradiological hazard would result from the chemical processing, handling, and transportation activities, during which accidents could happen.

The uncertainties associated with this unproven technology make precise analyses of these hazards difficult. Health evaluations, however, suggest that such hazards would pose approximately 20 times the risk of the radiological occupational hazards (Blomeke et al. 1980).

Other factors, such as seismic activity, fires, or severe meteorologic conditions, could lead to abnormal conditions. No such factors or their ensuing impacts, however, have been identified as warranting detailed environmental analysis for the transmutation facilities.

Natural System Impacts

Transmutation activity would involve handling several chemicals posing a potential health hazard. These chemicals would represent a threat to the natural environment surrounding fuel handling and processing facilities, as well as to the interconnecting transportation networks. Individual impact scenarios have not been postulated, but it can be assumed that there would be a risk of nonradiological impact associated with use of these chemicals not unlike that experienced by certain chemical process industries today.

Other nonradiological ecosystem impacts would result from construction, operation, and maintenance activities. Such impacts cannot be fully addressed except for a specific site. In general, potential impact would be similar to that of a comparably sized industrial operation. Reductions in the quantities of natural vegetation, an increase in runoff, and elimination of certain habitats are types of impacts that would be expected from such a facility. Although similar to impacts described for the baseline case of a fuel reprocessing operation that includes a mined geologic repository, the transmutation impacts would be greater because additional facilities and increased transportation would be involved.

Socioeconomic Impacts

Socioeconomic impacts associated with the transmutation alternative would occur primarily as a result of construction, operation, and transportation activities. Implementation of this alternative would involve a major construction force of over 3,000 individuals. Employment needs during operation would diminish to approximately 350 individuals per year for the FRP-WTF and 250 for the FFP-WTE (Smith and Davis 1980). These activities would also support increased transportation employment.

Compared to the baseline case of reprocessing without partitioning, operational employment levels for transmutation would increase substantially at the reprocessing and MOX fuel fabrication centers. Estimated work force increases are 35 and 80 percent at reprocessing and fuel fabrication facilities, respectively. Estimated socioeconomic impacts of such facilities are only conjectural at this point and specific impacts of hypothetical communities and groups are not included in this discussion.

Aesthetic Impacts

No data exist suggesting that aesthetic concerns from facilities required for transmutation activities would be greater than those associated with the reprocessing without partitioning. Neither the appearance or noise levels produced from the additional partitioning facilities should vary significantly from the baseline fuel reprocessing and preparation facilities.

Resource Consumption

Fuel and raw materials used in construction, as well as the chemicals and fuel required during operations and subsequent transportation activities, would be the most important resources used in the partitioning and transmutation process. For construction activities, a range of energy sources would be used in hardware fabrication and in actual construction operations. Other building materials such as steel, sand, and gravel typically used in major construction activities would also be consumed.

The reprocessing and partitioning process would also require quantities of chemicals, including nitric acid, hydrofluoric acid, hexanitrate acid, and several solvents. These chemicals would react with the waste material to form secondary wastes, as well as the desired end products.

Additional land would be required for this alternative. Facilities at the reprocessing plant should occupy 70 ha (172 acres) (Smith and Davis 1980) compared with 36 ha (90 acres) at present (DOE 1979c), and at the fuel fabrication plant 24 ha (59 acres) (Smith and Davis 1980) compared with 3 ha (8 acres) at present (DOE 1979c). Such a facility would normally process approximately 400 MTHM/year. In addition to the acreage occupied by each facility, large "restricted" areas would have to be established. Because of the conceptual nature of these facilities and the many possible ways they might be laid out, there are no specific estimates of the total size of restricted areas. At a minimum, the combined reprocessing and waste treatment facility would require a 2400 ha (6000-acre) restricted area while the fuel fabrication plant would require a 4000-ha (10,000-acre) restricted area. These figures are based on estimates for the reprocessing and fuel fabrication plants without waste treatment facilities (DOE 1979c).

International and Domestic Legal and Institutional Considerations

The primary institutional concern associated with implementation of a transmutation process would be the compatibility between such a system and existing power reactors. Specifically, the use of commercial power reactors as transmutation devices might result in significant fissile penalties, reactor peaking problems, reduced reactor availability, shielding requirements for fresh fuel, increased operating costs, and the need for significantly more enriched ^{235}U as a driver fuel. Consequently, technological improvements in transmutation processes or an evaluation of the institutional framework surrounding establishment of new nuclear plant operating standards is needed before the transmutation alternative can be implemented.

Finally, it must be recognized that the partitioning and transmutation processes include intensive reprocessing of nuclear waste material and plutonium recycle. Adoption of the transmutation alternative therefore, would be inconsistent with this nation's current policy regarding reprocessing.

6.1.7.5 Potential Impacts Over the Long Term (Postemplacement)

Successful implementation of the transmutation process would reduce the long-term hazards associated with waste material. In fact, effective transmutation would virtually eliminate concerns with actinides and their daughters. Although the potential long-term benefits would be significant, there are long-term uncertainties and problems that must be weighed against them.

Potential Events

For this option, TRU-depleted wastes are assumed to be sent to a mined geologic repository. Therefore, events leading to potential problems over the long term for this option would be the same as those associated with the mined geologic repository (see Section 5.6). A major difference exists in impacts, however, because transmutation wastes would not be as toxic in the long term (beyond 1,000 years).

Potential Impacts

Impacts over the long term would be expected to be less severe than those anticipated with reprocessing only, since the waste placed in the repository would be partitioned and transmuted to reduce its toxicity. An important exception to this would occur following early termination of the transmutation cycle. Such termination can actually increase the radiotoxicity of the wastes, as mentioned earlier (Croff et al. 1977).

Results of a long-term risk comparison (Logan et al. 1980) between a reference (no transmutation) and a transmutation fuel cycle indicate that:

- Cs-137 and Sr-90 would dominate the health effects during the first few hundred years for both fuel cycles.
- After a few hundred years and for several tens of thousands of years thereafter, the most significant nuclides for the reference fuel cycle would include a generous mix of actinides and their daughters at a significantly reduced activity level. Transmutation would strongly reduce the effects during this period.
- During later years, two nuclides, Tc-99 and I-129, which are released by leaching, would completely dominate all other nuclide contributions. Because these nuclides are not removed through transmutation, the results show no benefit during these later years.

Long-term health effects have been integrated over 1 million years to determine the long-term probabilistic (expected) risk (Blomeke et al. 1980 and Logan et al. 1980). The long-term risk was found to be controlled to a very large extent by the contributions from Tc-99 and I-129, which constitute about 99 percent of the integrated risk. This is because (1) the slow leach incident dominates the long-term probabilistic risk since it was assumed to have a much higher probability of occurrence than a volcanic or meteor incident and (2) only those nuclides that sorb poorly or not at all (i.e., iodine, technetium, carbon) migrate through the geosphere quickly enough to reach the biosphere within 1 million years. Therefore, transmutation of actinides would have its most substantial value if an unlikely event occurs. For example, the probability of a volcanic incident is only one in 100 billion, but if it should occur, the radioactive material could enter the biosphere very rapidly.

Looking at the issue described above in another way, it is noteworthy that catastrophic events occurring beyond 100 years following emplacement would not cause significant radiologic health effects if transmutation where employed.

6.1.7.6 Cost Analysis

The cost of utilizing transmutation to modify the radionuclide composition of waste would be added to the cost of disposal associated with remaining modified waste. However, modification of the waste's radionuclide content has the potential to alleviate some of the disposal requirements and reduce these costs. Such costs have not been developed at this time.

Costs have been developed for a fuel cycle including actinide transmutation utilizing commercial light water reactors as the transmutation device. These were compared with the costs of a mixed-oxide fuel cycle (Alexander and Croff 1980). This study indicated cost increase of about 3 percent for nuclear generated electricity if actinide transmutation were utilized for disposal purposes.

The significant cost differentials were associated with the requirement of specialized partitioning facilities and hardware. The continued recycle of actinides into the fuel cycle would increase the neutron activity within the fuel material about tenfold for spent fuel and more than 100 times for fresh fuel. These increases must be taken into account by increased shielding and by use of remote operations and maintenance when designing fuel cycle facilities. Reprocessing costs would increase by an estimated 5 percent, fuel fabrication costs would double, and transportation costs would nearly triple (Smith and Davis 1980).

The following cost estimates are for only the specialized partitioning facilities colocated with their respective mixed-oxide fuel fabrication facility and spent fuel reprocessing facility. The fuel fabrication plant has a throughput of 660 MTHM per year and the reprocessing plant a throughput of 2,000 MTHM per year.

Capital Costs

The partitioning process buildings are first-of-a-kind facilities that, in several instances, include process operations that have not advanced beyond laboratory test and evaluation. Therefore, considerable judgment was used in the development of the capital costs shown in Table 6.1.27.

Operatings Costs

Estimated operating costs are shown in Table 6.1.28. Labor cost estimates are based on an average salary of $20,000 per year for management, engineering, and supervision and $14,500 per year for operators, maintenance personnel, guards, laboratory technicians, and clerical personnel.

TABLE 6.1.27. Capital Costs For Partitioning Facilities (Millions of 1978 Dollars) (Smith and Davis 1978)

	Colocated With Reprocessing Plant			Colocated With Fuel Fabrication Plant		
	Material	Labor	Total	Material	Labor	Total
Land Improvements	1.3	1.2	2.5	1.0	.9	1.9
Process Facilities	200.0	127.0	327.0	73.1	46.9	120.0
Tunnel and Piping	5.8	10.6	16.4	4.9	9.2	14.1
Support Facilities	13.0	5.7	18.7	12.2	4.6	16.8
Subtotal	220.	145	365.	91	62	153
Field Indirects and S/C's OH&P			145			62
Subtotal			510			215
Engineering & Design			143			60
Subtotal			653			275
Contingency			228			96
Total			881			371

TABLE 6.1.28. Operating Costs For Partitioning Facilities (Millions of 1980 Dollars)

	Colocated With Reprocessing Plant	Colocated With Fuel Fabrication Plant
Process Chemicals	16.0	1.4
Utilities	6.2	2.2
Labor	8.2	5.8
Equipment Replacement	3.8	1.0
Property Tax and Insurance	26.0	11.1
NRC License and Inspection	0.2	0.2
Total	60.4	21.7

Decommissioning

Decommissioning costs associated with the partitioning facilities were estimated to be 12 percent of the capital costs for the partitioning facilities, i.e., $105 million for the facility colocated with the reprocessing plant or $45 million for the facility colocated with the fuel fabrication plant.

6.1.7.7. Safeguard Requirements

The transmutation concept depends on processing of the spent fuel elements and the recycle of transmutable materials. The extra processing and transportation, and the availability of sensitive materials at all points in the back end of the fuel cycle would increase the opportunity for diversion of these materials. In addition, because of the necessity to process and recycle material eight or nine times to ensure full transmutation, the annual throughput of sensitive materials would greatly increase. Material accountability would also be more difficult because of the large quantities and high irradiation levels. Safeguards of recycled plutonium would be simplified because of the higher concentration of ^{238}Pu. Also, recycled actinides containing ^{252}Cf and ^{245}Cm would require shielding from neutrons that should simplify safeguard requirements. Furthermore, because geologic disposal would be required on the same scale as discussed in Chapter 5, all the safeguard requirements described there would also be required for a fuel cycle using transmutation. See Section 4.10 for additional discussion of predisposal operation safeguard requirements.

6.1.8 Space Disposal

6.1.8.1 Concept Summary

Space disposal offers the option of permanently removing part of the nuclear wastes from the Earth's environment. In this concept, HLW would be formed into a cermet matrix and packaged in special flight containers for insertion into a solar orbit, where it would remain for at least 1 million years. NASA has studied several space disposal options since the early 1970s. A reference concept using an uprated Space Shuttle has emerged and is considered in detail here.

The Space Shuttle would carry the waste package to a low-earth orbit. A transfer vehicle would then spearate from the Shuttle to place the waste package and another propulsion stage into an earth escape trajectory. The transfer vehicle would return to the Shuttle while the remaining rocket stage inserted the waste into a solar orbit.

The space disposal option appears feasible for selected long-lived waste fractions, or even for the total amount of high-level waste that will be produced. The remaining TRU wastes would require some terestrial disposal option, such as mined geological repositories in the continental U.S. Space disposal of unreprocessed fuel rods does not appear economically feasible or practical because of the large number of flights involved.

Space disposal was considered for its potential to reduce long-term environmental impacts and human health effects for a given quantity and type of waste compared with alternative terrestrial disposal options. Because of the characteristics of the space disposal concept, which removes the waste package from the bioshpere, it is highly unlikely that physical forces would cause the radioisotopes to migrate toward the Earth. Consequently, for a package properly placed in orbit, there would be no long-term risk or surveillance problem as in terrestrial alternatives. However, the risk and consequence of launch pad accident and low earth orbit failure must be compared to the risk of breach of deep geologic repositories.

6.1.8.2 System and Facility Descriptions

System Options

The reference concept and system for the initial space disposal of nuclear waste has been developed from a number of options available at each step from the reactor to ultimate space disposal. These options are summarized in Figure 6.1.22 (Battelle 1980), which indicates currently preferred options chosen for the DOE/NASA concept, primary alternatives, secondary alternatives, and options that are no longer considered viable. The bases for selection of options for the reference concept (those blocked off) are detailed in various sources listed in Appendix M.

Waste-Type Compatibility

As noted, space disposal of unreprocessed spent fuel rods would be impractical because an excessive number of launches would be required. This would result in high energy re-

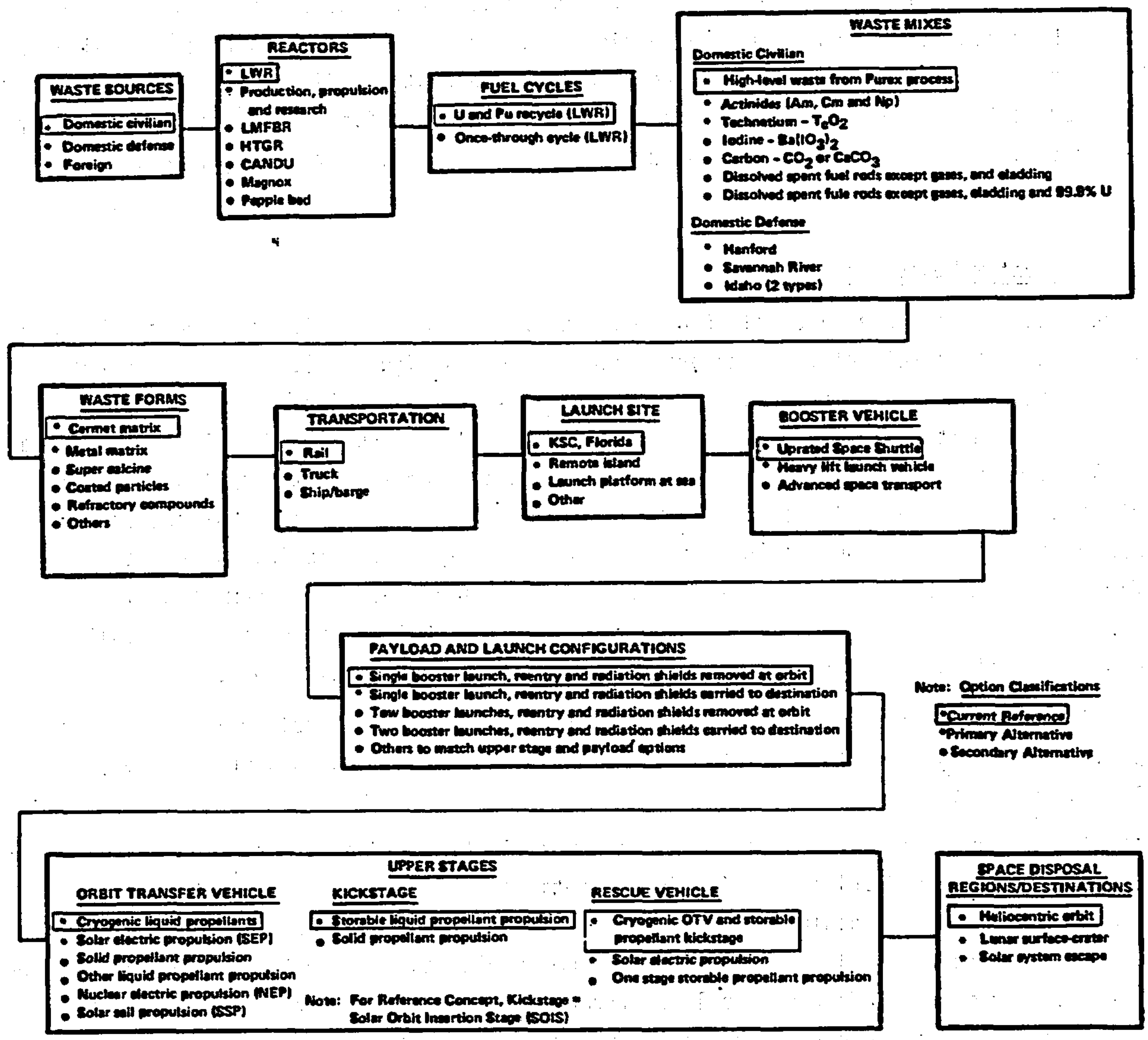


FIGURE 6.1.22. Major Options for Space Disposal of Nuclear Waste

quirements, high costs, and probably increased environmental impacts (see Section 6.1.8.4). Thus, some form of waste separation would be required. For HLW, the option appears to be feasible, on the basis of the much lower number of Space Shuttle flights that would be required (approximately one launch per week to dispose of HLW from 5000 MT of heavy metal resulting from operations of approximately 170 GWe nuclear capacity). It is also possible that the space option would be used to rid the Earth of smaller quantities of radioactive wastes that pose special hazards for long-term terrestrial disposal. The disposal of selected isotopes would require chemical partitioning, with its high costs and secondary waste streams. Remotely handled and contact-handled TRU wastes from the recycle options would require geologic disposal.

Waste-System Description

The concept for space disposal of nuclear waste described here is the current DOE/NASA reference concept as relfected by the preferred options in Figure 6.1.22. To place the space disposal concept into perspective from a total system viewpoint, Figure 6.1.23 shows the waste management system, emphasizing the location and process flow details of the space disposal alternative within the total system. Two points are apparent from this figure: (1) chemical processing would definitely be required for space disposal of waste, and (2) the mined geologic repository would be part of the total system. The following discussion briefly summarizes the mission profile frm the standpoint of waste-type compatability, prelaunch activities, and orbital operations. Battelle (1980) presented a more detailed discussion of this profile and various element definitions and requirements.

Prelaunch Activities. The prelaunch activities would include nuclear waste processing and payload fabrication, ground transportation of waste, on-site payload preparation, and final staging operations.

Typically, spent fuel rods from domestic power plants would be transported to the waste processing and payload fabrication site in conventional shipping casks (see Chapter 4). A high-level waste stream containing fission products and actinides, including several tenths of a percent of the original plutonium and uranium, would result from the uranium and plutonium recovery process. This waste would be formed into a "cermet" matrix (Aaron et al. 1979) (an abbreviation for ceramic particles uniformly dispersed within a metallic phase), which has been shown to have superior properties compared with other potential waste forms for space disposal (Battelle 1980). The waste would then be fabricated into an unshielded 5000-kg sphere. Within a remote shielded cell, this waste payload would be loaded into a container, which would be closed be sealed, inspected, decontaminated, and packaged into a flight-weight gamma radiation shield assembly. During these operations and subsequent interim storage at the processing site, the waste package would be cooled by an auxiliary cooling system.

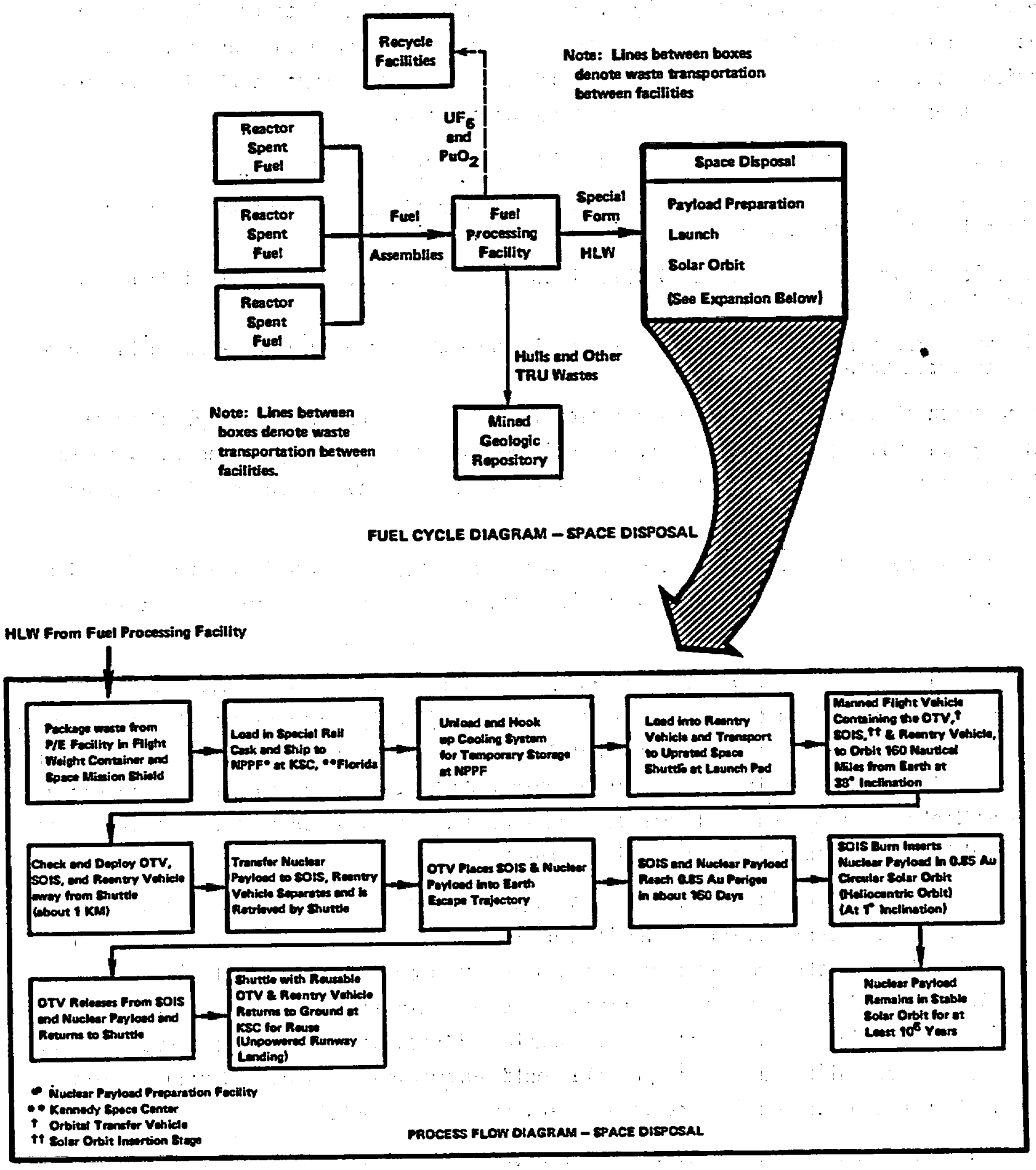


FIGURE 6.1.23. Waste Management System--Space Disposal

The shielded waste container would be loaded into a ground transportation shipping cask. This cask would provide additional radiation shielding, as well as thermal and impact protection for the waste container to comply with NRC/DOT shipping regulations. It would be transported to the launch site on a special rail car and be stored in a nuclear payload preparation facility with provision for additional shielding and thermal control. The waste containers would be monitored and inspected during storage.

For launch, the shielded waste form would be integrated with:

- A reentry vehicle, which would protect and structurally support the waste in the Space Shuttle orbiter cargo bay
- A solar orbit insertion stage (SOIS), which would place the waste payload into its final solar orbit
- An orbit transfer vehicle (OTV), which would take the waste from low Earth orbit into a solar orbit transfer trajectory.

Prelaunch checkout would include verification of the payload and the payload-to-orbiter interface systems. Typically, propellant would be loaded in the preparation facility to minimize the hazard of propellant loading while the payload was in the Shuttle cargo bay on the launch pad.

From the preparation facility, a special-purpose transporter would take the payload to the launch pad, where special equipment would position and install it in the Shuttle cargo bay.

Orbital Operations. The orbital operations for this concept would include launching into earth orbit, transfer from there to a solar orbit, and finally rounding out the solar orbit. (see Figure 6.1.24). The Uprated Space Shuttle, designed to carry a 45,000 kg (99,000 lb) payload, would be launched into a low Earth orbit (300 km). The launch would avoid early land overflight of populated land masses. The liquid rocket booster engines and the external tank would be jettisoned before the orbit is reached.

During suborbital portions of the flight, the Orbiter would be able to command shutdown of all engines and either return to the launch site or ditch in the ocean. From 5 to 6 minutes after launch, the Orbiter could abort by going once around the Earth and then returning to land. After 6 minutes, the Orbiter has the on-board thrust capability to abort directly to a sustained earth orbit. If a Shuttle malfunction exceeded the abort capability, the nuclear payload with the reentry vehicle would automatically eject and make its own reentry. It would be designed to survive a land or water impact.

Once in orbit, the loaded reentry vehicle would be automatically latched to the SOIS and, with the OTV, would automatically deploy from the orbiter bay. At this time, the waste payload would be remotely transferred from the reentry vehicle to the SOIS payload adapter.

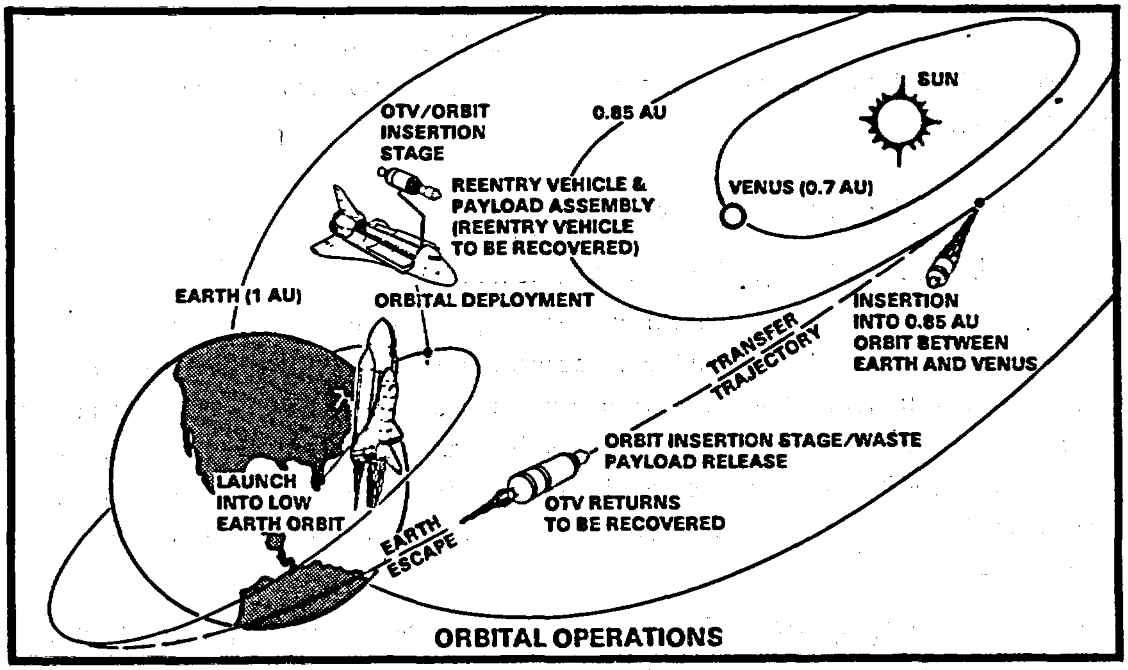


FIGURE 6.1.24. Orbital Operations

After a final systems checkout, the OTV would place the SOIS and its attached waste payload into an Earth escape trajectory. Propulsion would be controlled from the Orbiter, with backup provided by a ground control station. After propulsion, the OTV would release the SOIS/waste payload and would return to low Earth orbit for rendezvous with the Orbiter. The payload would require about 163 days to reach its perihelion at 0.85 astronomical units (A.U.) about the Sun. (One A.U. is equal to the average distance from the Earth to the Sun.) Calculations have shown that this orbit would be stable with respect to Earth and Venus for at least 1 million years.

In case of OTV ignition failure, a rescue OTV would be launched to meet and dock with the SOIS for propulsion into the escape trajectory. Safety features would be included in the design of this vehicle to prevent reentry of the unshielded payload into the Earth's atmosphere (Bechtel 1979a).

After rendezvous with the OTV, the Shuttle Orbiter would return to the launch site for refurbishment and use on a later flight. The empty reentry vehicle would also be recovered and returned with the Shuttle for reuse. The normal elapsed time from launch to return to the launch site would be 48 hours (Bechtel 1979a).

Systems for tracking the vehicles during launch, earth orbit, and the earth escape trajectory exist. There is also a system for locating and tracking the payload in deep space at any future time. However, once the proper disposal orbit had been verified, no additional tracking should be necessary.

Retrievability/Recovery. Until the waste package had been successfully disposed of in accordance with the design, retrieval or recovery capability would be necessary. A discussion of the rescue technology required for such a retrieval capability is presented in Section 6.1.8.3 below.

6.1.8.3. Status of Technical Development and R&D Needs

Present State of Development and Technological Issues

While the space option appears technically feasible, there are engineering problems that would require resolution. The Space Shuttle is currently in development and the first orbital flight is scheduled in 1981. The Space Transportation System should eventually (1990s) include a Space Shuttle with liquid rocket boosters (replacing current solid rocket boosters) and a reusable OTV. NASA has studied such vehicles extensively for future space missions and they represent a logical extension of the space transportation capability upon which to base a reference concept.

Many aspects of the space disposal system represent straightforward, applications of existing technology, e.g., use of liquid propellants and reentry vehicle design; however extensive engineering development would be required. The major technology development requirements are in design for safety, environmental impact analysis of space launches, and waste preparation. The nuclear waste payload container and reentry vehicle are only conceptually defined and additional study would be required to assure that safety and environmental requirements could be met in case of launch pad and reentry accidents. Development of a capability for deep space rendezvous and docking to correct improper orbit of a waste package would be required. The current status of development and research needs in specific areas are discussed below.

Emplacement Methods. The technology for launching both nuclear and nonnuclear payloads into space is highly developed, but the technology for putting nuclear waste in space is still in a conceptual stage. Earlier experience with space nuclear auxiliary power (SNAP) systems employing radioactive thermoelectric generators provides some experience, particularly in safety analyses, but the amounts of radioactive materials in such systems are much less than those that would be associated with waste payloads. The present DOE/NASA conceptual definition is based on technology and equipment used previously in other space missions but which would require design modifications for use in waste disposal missions. For example, the Space Shuttle power plant would need to be upgraded to increase payload capacity and thereby reduce the number of flights required. On the basis of the results obtained in the space program, considerable confidence has been gained in ability to design the necessary high-reliability systems. Procedures currently being developed to address abort contingencies for the manned Space Shuttle would be useful to mitigate adverse effects of aborts in waste launch operations.

Waste Form. The waste form would have to be a nondispersible, chemically stable solid. The composition of this waste has not been defined by the space program sponsors, but there are several possible candidate processes that might produce the proper form, as suggested in Figure 6.1.22.

The waste form should contribute to overall system safety, especially for potential accident sequences, and should also contribute to system optimization in terms of payload, economics, and materials compatibility. Desirable attributes are:

- High HLW to inert content ratio
- High thermal conductivity
- Resistance to thermal shock
- Thermochemical stability
- Toughness
- Low leachability
- Applicable to both commercial and defense wastes
- Resistance to oxidation
- Low cost
- Ease of fabrication.

Because weight would be important in the launching operation, the waste forms should also maximize the amount of waste carried at each launch (waste loading). An iron/nickel-based cermet prepared by ORNL for other disposal options appears suitable, but would require further development.

Waste Package. The reference waste package would consist of the spherical waste form surrounded by a metal cladding, a gamma shield, a steel honeycomb structure (for impact), insulation (for reentry), a graphite shield (for reentry), and the reentry vehicle itself, which would contain the waste during launch and Earth orbit in case of accident. Only conceptual definitions have been developed.

Waste Partitioning. Certain space option alternative concepts would be enhanced if specific isotopes were removed from the waste, e.g., strontium or cesium. Alternatively, space disposal might be more appropriate for certain species, e.g., iodine, technetium, the actinides, or all three. Technology development would be needed to provide these partitioning options.

Facilities. The size, capacity, and functional requirements of the nuclear payload preparation facility are not defined. Major design tasks remain before this facility could be developed.

Rescue Technology. Remote automated rendezvous and docking capabilities would probably be required for space disposal of radioactive waste. The HLW payload would require technology development to provide recovery capabilities for payloads in deep space, especially for uncontrollable and/or tumbling payloads. Also, it might be necessary to develop new technology for deep ocean recovery of aborted or reentrant payloads. Deep ocean recovery has been demonstrated on several recent projects, but any new, special capabilities to handle HLW payloads would need to be defined. Special equipment to recover reentrant payloads that touch down on land might also be required, although the technological challenge would probably not be as great.

R&D Requirements

In the final analysis, R&D needs would depend on the space disposal mission selected. The R&D requirements for this program would span the spectrum from systems definition conceptual studies through generic technology development (e.g., waste form) to engineering developments of facilities and hardware (e.g., the payload preparation facility and tailored space vehicles). These latter aspects would be deferred until the space disposal mission is better defined.

Thus, initial R&D would need to cover the following elements for concept definition and evaluations, listed approximately in sequential order.

- Perform trade-off and risk analysis studies to select the mix of radionuclides for space disposal
- Assess technology availability of waste processing and waste partitioning options
- Develop waste form criteria and options for space disposal
- Define facilities and ground transportation systems requiring R&D
- Define waste payload systems and containment requirements
- Define and select flight support systems for the space disposal option (e.g., shielding)
- Complete conceptual definition of unique launch site systems
- Assess advanced launch systems under development for space disposal applicability
- Define possible systems for transferring nuclear waste from Earth orbit and recovering failed payloads
- Characterize possible space destinations and missions
- Assess unique safety and environmental aspects of the space mission (e.g., launch pad fires and explosions affecting the waste package).

These conceptual studies would set the requirements for future R&D programs, if warranted. Other applicable ongoing R&D projects, e.g., concept definition of metal matrix waste forms and advanced launch system definition, would be pursued concurrently.

Implementation Time

With the space disposal mission currently in the concept definition and evaluation phase, meaningful predictions of the initial operational date are not possible. However, the present DOE/NASA concept depends on the availability of an OTV and the Uprated Space Shuttle that have not been developed. This space disposal system could be operational possibly by the year 2000. Major sequential outputs that could be derived from conceptual studies are:

- Identification of viable alternative space systems concepts
- Identification of viable nuclear waste system concepts
- Selection of preferred concepts
- Selection of baseline concept
- Completion of baseline concept definition
- Generation of development plan

Estimated Development Costs

Development costs would depend largely on the specific space option approved. Also, once that option was defined, ongoing work oriented to other Shuttle and waste disposal options could be refocused on space disposal requirements. Examples are deep space rendezvous and docking techniques and waste form technology development. This would identify the incremental Shuttle and waste isolation program costs attributable to space disposal.

Thus, funding requirements for development of the space disposal option have not been well defined. It would generally be assumed that NASA would undertake the development of the required space components and DOE would develop the waste technology if the concept was pursued. It assumed that the approach would be on an incremental basis. This work would include R&D and identification of design development requirements for nuclear waste systems and space systems for disposal, domestic/international affairs studies, and impact assessments. The studies would provide a cost basis for further programmatic decision making.

Summary

Major uncertainties, shortcomings, and advantages of the concept are summarized below:

- The concept does not permit ready corrective action.
- The concept is susceptible to single mode (launch pad) failure, unless well-engineered multiple barriers are developed to protect the waste.
- Significant technology advances and equipment development will be required.
- Waste form and package concept development are in a very preliminary stage.
- The concepts usefulness would be limited to waste from reprocessing or further limited to selected isotopes.

6.1.8.4 Impacts of Construction and Operation (Preemplacement)

A space disposal approach must consider the total integrated system risk, i.e., the risks of launching wastes into space and the risks associated with the secondary waste streams generated by waste treatment, the fraction of waste that would have to go to terrestrial disposal, and the increase in system complexity. Hence, the short-term health and environmental impacts would likely be increased, while risks associated with those residual waste forms that remained on Earth for disposal in a mined geologic repository would likely be decreased. The environmental and health impacts associated with the latter consideration are expected to be less significant than those associated with total terrestrial disposal of HLW.

In the early years of a space disposal program, certain modifications would be required at Kennedy Space Center, assuming it was selected as the launch site. At the least, this would involve construction of a payload preparation facility. If the total Space Shuttle traffic (including all space missions) saturated the capability of shuttle facilities, then modifications, or even new facilities (e.g., launch pads), would be necessary. New construction activities would be designed to have the minimum adverse effect on the area. NASA has concluded that all potential nonradiological environmental impacts foreseen during normal operation of the Space Shuttle would be localized, brief, controllable, and of minimum severity (NASA 1978). Results of an evaluation of the incremental impacts of construction of facilities to accommodate waste disposal via the Shuttle and other environmental impacts of the space disposal program are presented below (Bechtel 1979a).

Health Impacts

Normal operation of facilities are not expected to cause any significant adverse health effects from either radiological or nonradiological sources. During abnormal operations (a reentry and burnup accident) the total population radiological dose could be quite large; although the estimated average individual dose would be very small.

Radiological Impacts. Health impacts from routine operations would be related primarily to planned release of radioactive and nonradioactive materials. Impacts to man from routine operations would be derived from three of the five operational phases: predisposal treatment and packaging (reprocessing), transportation, and emplacement.

No significant adverse health effects would be expected from normal operation of reprocessing facilities (NRC 1976). Incremental effects of additional processing to partition specific nuclides are not expected to change this conclusion.

Health effects caused by terrestrial transportation would be expected to be no different for space disposal than for other waste disposal options and are assumed to be similar to those for existing containers that have been reviewed for safety and licensed by regulatory agencies.

The estimated total occupational whole-body radiation dose from space disposal (the three operational phases plus the terrestrial repository for secondary waste) is 6340 man-rem/yr

(Bechtel 1979a). (See Table 6.1.30.) Of this dose, 1000 man-rem/yr derives from Space Shuttle-related activities. The nonoccupational dose is estimated at 180 man-rem/yr, with a negligible amount attributed to the Space Shuttle program.

Accidents may be classified by their location within the sequence of operations as associated with:

- Waste treatment
- Payload fabrication
- Payload ground transportation
- Handling and launch preparation
- Launch phases (suborbital)
- Orbital operations
- Postemplacement.

Within this sequence, many possible accidents that might be called "typical industrial" accidents can be identified. These are not discussed further because they (a) are not related directly to either the nuclear or space transportation aspects, (b) have negligible environmental impact, and (c) are no more probable (and in fact may be less probable) in this activity than in any industrial activity of similar magnitude. Of primary concern here are those accidents involving radioactive material, that would lead to the release and dispersion of the radioactive material into the environment. Waste treatment, payload fabrication, payload ground transportation and handling, and launch preparation for the space disposal option would be expected to be broadly similar to the same activities as employed for terrestrial disposal options. Thus, the possible accidents and accident consequences would also be similar (subject to some variation relating to the different nuclides that might be involved). Such accidents and their consequences are treated in Chapter 4 and are not further described here.

Certain types of accidents that might occur during the launch or orbital and postemplacement operations would impose difficult environmental conditions on the payload. They could lead to the payload coming to rest in uncontrolled areas or to the release and dispersion of some of the radioactive waste. These accident types would include:

- Explosions
- Intense fires
- High-velocity impact
- Atmospheric reentry.

The payload and other mission hardware, as well as the procedures used to carry out the various operations, would be designed to

TABLE 6.1.30 Short Term (Preemplacement) Radiological Impacts For The Space Disposal Program Normal Operation

	Whole-Body Dose, man-rem/yr	
	Occupational	Nonoccupational
Waste Processing Facility	4100	90
Transportation	210	90
Repository (Secondary Waste)	1030	Neg.
Space		Neg.
NPPF	70	
Transporter/Launch Pad	150	
Shuttle	780	
	6340	180

- Minimize the probability of events leading to severe environments
- Provide, when possible, a contingency action to remove the payload from the threatening environment
- Maximize the probability that the waste payload containment will not be violated if subjected to the environment.

Two important types of accidents, both unique to the space disposal option, are:

- A catastrophic, on- or near-pad explosion and fire of the booster launch vehicle
- A high-altitude reentry and burnup of an unprotected nuclear waste container, with subsequent conversion of a certain fraction of the payload to submicron particles of metal oxides.

Aside from immediate possible casualties and the close-in physical effects from, for example, the on-pad explosion and fire, the environmental impact of overriding significance for these events would be possible radiation exposure to the general public. Edgecombe et al. (1978) provides preliminary data on environmental conditions around catastrophic launch-pad accidents.

Short-term risks might or might not be lower than those for terrestrial disposal options. However, for the space disposal option to be implemented, they would have to be at an acceptable level. Reliability data for systems would be required before a risk assessment could be made. Reliabilities of the booster vehicle, upper stages, and safety systems envisioned for the space disposal mission have not yet been determined by NASA, but are expected to be high.

Regarding on- or near-pad accidents, no precise estimates of health effects from worst-case credible accidents can be made from present information. Nonetheless, dose commitments to the most exposed individual (80 rem/event) and to the population within 100 km of the site (4000 man-rem/ event) have been estimated for the on-pad accident (Bechtel 1979a). More work would be needed concerning the integrity of the nuclear waste container systems that would be employed for the space disposal option and the actual accident environments that would result. Additionally, the relationship between shielding and possible health effects during recovery from major accidents would require further technical study. Under accident conditions, however, the stability of the HLW is expected to reduce the consequences of any loss of containment (DOE 1979a).

In a space disposal reentry and burnup accident, the estimated average and individual dose is "quite small", yet the total population dose could be very large (e.g., about 10^7 man-rem/accident to the world population) (Bechtel 1979a).

Nonradiological Impacts. Generally, environmental impacts that would be caused by normal operations or nonradiological-type accidents from a space disposal option are not expected to be significant (NASA 1978). Potential environmental impacts related to the normal operations of space transportation systems that might be unique are discussed below.

The types of environmental health impacts that could be attributed to normal space transportation activities are:

- Gaseous and particulate emissions from rocket engines
- Noise generated during launches and landings (including sonic booms)
- Commitments of nonrecoverable resources.

These effects have been studied by NASA and an environmental impact statement has been issued (NASA 1978). To date, research has indicated there would be no significant effects to the human population from a steady launch rate of 60 shuttle flights per year.

During abnormal conditions, the major nonradiological concern appears to be whether or not large pieces of metal would reach the ground in the event of an upper stage failure. This question and others are the subject of ongoing investigations.

Natural System Impacts

Radiological and nonradiological impacts are analyzed below for the natural system.

Radiological Impacts. Environmental studies of the Barnwell Nuclear Fuel Plant (AGNS 1971, 1974; Darr and Murbach 1977) provide information concerning environmental impacts expected from normal processing of the reference waste mix. Expected environmental effects include modest heat additions to local water systems, as well as both gaseous and liquid releases of radioactive and nonradioactive materials.

In general, normal operation within regulatory limits should assure that ecosystem radiological impacts are acceptable. These conclusions are confirmed by generic studies (DOE 1979b).

The data base for environmental assessment of the space option is very preliminary at this time. Environmental assessments could be made only when the total system has been better defined. Bechtel (1979a) provides a recommended schedule for assessing ecosystem impacts from abnormal events, which, if adhered to, would make preliminary results available late in 1980.

Nonradiological Impacts. The major environmental impacts from construction of required waste treatment, payload fabrication, payload receiving, and launching facilities would be qualitatively similar to those of other construction activities. Construction impacts, in general, are related to resource commitments (land, water, and materials) and to effects on environmental quality and biotic communities from the pollutants and fugitive dust released by construction activities.

Water quality would be adversely affected by the creation of sedimentation resulting from runoff at construction sites, discharge of treated wastewaters and blowdown at reprocessing facilities, and salt pile runoff at the secondary waste repository (Bechtel 1979a).

Air quality during construction would be adversely affected as a result of fugitive dust and diesel equipment emissions, emissions from waste and employee transportation, and salt drift (Bechtel 1979a). On the basis of results of analyses performed for air quality, water quality, land quality, weather, and ecology during normal operations, no long-term or cumulative effects are predicted for the abiotic and biotic communities (NASA 1978).

Accidents related to Space Shuttle launches (without payloads) have been described elsewhere (NASA 1978) and are not expected to be environmentally significant.

Socioeconomic Impacts

Manpower estimates for construction and operation are a key variable in assessing socioeconomic impacts. Employment related to payload handling and launch is a differentiating factor between the space option and other waste disposal options.

Only preliminary data for the socioeconomic assessment of the space option are available at this time. A detailed assessment of the socioeconomic implications of the space disposal option would require more accurate employment estimates, information on the industrial sectors affected by capital expenditures, and identification of the specific geographic areas involved. Rochlin et al. (1976) provide a general discussion of the socioeconomic implications of nuclear waste disposal in space.

(a) While Kennedy Space Flight Center has already adjusted to many of the impacts mentioned below, selection of an alternative launch site would require additional impact assessment.

- Public Sector Economy. Current estimates of launch rates suggest that support of the entire space transportation system for the space disposal activity might require 25,000 to 75,000 employees. This work force represents a substantial payroll and a large number of households throughout the country that would constitute sizable demands for goods and services. The environmental impact statement for the Space Shuttle (NASA 1978) provides insight as to where money would be spent.

- Private Sector Economy. In addition to direct employment, the space disposal option would induce secondary employment, as well as major capital investment. This additional economic activity would, in turn, generate additional demands for goods and services.

- Population Size and Growth Rate/Population Composition. The size and geographic distribution of the work force levels would affect the magnitude and location of the socioeconomic impacts. The ability of local areas to meet such demands will affect the severity with which these impacts are perceived. Greater project definition and detail are necessary before these impacts can be accurately assessed.

Aesthetic Impacts

Aesthetic impacts for those aspects of the program unique to space disposal would be generally limited to noise and visual features.

Noise. Only the Orbiter reentry would produce sonic boom over populated areas. Extensive studies of sonic boom dynamics indicate that the maximum effects would be at the nuisance or annoyance level (NASA 1978).

Appearance. Visual effects are expected to be significant because of the eight-story preparation facility and a 100-m stack for the reprocessing facility. Of course, actual site selection could have a mitigating effect on these impacts (Bechtel 1979a).

Resource Consumption

Launches of space vehicles always commit certain resources that are never recovered.

Energy. Estimated total energy requirements for the space disposal program (construction plus 40-year operation), which are considered significant, are summarized below (Bechtel 1979a).

Resource	Amount
Propane, m^3	1.0×10^7
Diesel fuel, m^3	1.5×10^6
Gasoline, m^3	1.3×10^5
Electricity, kWhr	5.9×10^{10}
Propellants, MT	
Liquid hydrogen	2.7×10^5
Liquid oxygen	3.7×10^6
Rocket propellant	7.2×10^5
Nitrogen tetroxide	2.4×10^4
Monomethyl hydrazine	2.0×10^4

Critical Resources. Estimated commitment of critical material resources required for construction plus 40 year operation (other than those required for launching) are characterized as follows (Bechtel 1979a).

Resource	Amount
Water, m^3	6.0×10^7
Steel and Major Alloys, MT	
Carbon Steel	2.9×10^5
Stainless Steel	3.0×10^4
Chromium	5.0×10^3
Nickel	2.0×10^3
Major Nonferrous Metals MT	
Copper	3.8×10^4
Lead	2.9×10^3
Zinc	6.0×10^2
Aluminum	8.3×10^4
Concrete, m^3	1.1×10^6
Lumber, m^3	4.0×10^5

Land. Approximately 9000 ha (22,230 acres) of land would be required for the space disposal program. There is sufficient land capacity at the Kennedy Space Center to meet this requirement (Bechtel 1979a).

International and Domestic Legal and Institutional Considerations

The space disposal option has elements that are unique and that would have to be addressed in a comprehensive analysis of this alternative. For example, careful assignment of responsibility and accountability will have to be made among the federal agencies that would be involved in this disposal option.

The space disposal option would also present international concerns that would have to be recognized and addressed. Potential issues are:

- Risk of accidents affecting the citizens of countries that did not participate in the waste disposal decision
- Possibility of joint disposal programs with other countries
- Assignment of associated costs to various countries.

In addition to these generic international issues, there are a number of specific multinational treaties, conventions, and agreements currently in force and subscribed to by the U.S. that bear upon the use of space for nuclear waste disposal. These include:

- "Treaty on Principles Governing the Activities of States in the Exploration and Use of Outer Space Including the Moon and Other Celestial Bodies" (1967)
- "Convention on International Liability for Damage Caused by Space Objects" (1972)
- "Agreement on the Rescue of Astronauts, the Return of Astronauts, and the Return of Objects Launched into Outer Space" (1972)
- "Convention on Damage Caused by Foreign Aircraft to Third Parties on the Surface" (1952)
- "Convention on Registration of Objects Launched into Outer Space" (1976).

This list suggests various issues that would have to be thoroughly explored in this early decision-making phase, including: (1) accident liability, (2) exclusive use of the lunar surface or other regions of outer space, and (3) international program involvement (e.g., use of the sea). These issues relate mainly to accident situations rather than routine operations.

In addition to these political and international issues, space disposal of nuclear waste would have a number of legal complexities associated with it, including liability and regulatory requirements (e.g., licensing). These concerns would be quite evident not only during, but also before and after actual implementation. Moreover, legal concerns could lengthen the time needed to implement a space disposal option.

6.1.8.5 Potential Impacts Over Long Term (Postemplacement)

Postemplacement for the space option is defined as the period of time after achievement of a stable solar orbit. Potential impacts during this period are analyzed for two different events: engineering failure and inadvertent human intrusion.

Potential Events

The possibility of sudden failure of a container in solar orbit would be extremely remote. However, if a container should rupture, for example, as a result of a meteor impact or degradation over the long term, the contents would be released and begin to spread. The physical processes by which the nuclear waste material would be dispersed in solar space include sputtering, thermal diffusion, and interactions with solar radiation and wind. Large pieces or particles of waste material would be sputtered into smaller particles, which in turn would disperse. The smallest particles, with radii less than 10^{-5} to 10^{-4} cm, would be swept out of the solar system by direct solar radiation pressure. Larger particles, those with radii up to 10^{-3} to 10^{-2} cm, would gradually loose momentum through scattering, charge exchange interactions, and collisions with energetic photons and solar wind protons. This process, called the Poynting Robertson effect, would cause these particles to begin moving in toward the sun where they would eventually be vaporized and broken down into smaller particles. Once this had occurred, the smaller particles would be swept out of the solar system by solar radiation pressure. This sweeping-out process would take an estimated 1000 to 10,000 years (Brandt 1970). NASA is currently studying this process.

The potential hazard from the isolated nuclear waste to persons on future space missions traversing the region about 0.85 A.U. is not known, but is believed to be extremely small and would be zero unless a manned trip by or to Venus were undertaken. Nuclear waste launched into an 0.85 A.U. orbit would not be recoverable for all practical purposes and the 0.85 A.U. solar orbit is far enough from the Earth and sufficiently stable that future Earth encounters would be effectively precluded (Friedlander et al. 1977).

Potential Impacts

With space disposal, waste would be isolated from the Earth for geologic time periods, in effect, permanently. Consequently, no long-term radiological or nonradiological health impacts are expected. The terrestrial component, storing only non-HLW, would therefore be minimized.

With regard to natural systems, upon retirement of waste processing fabrication and/or storage facilities (including the payload preparation facility), the land areas could be returned to other productive uses. Although details of decommissioning are not available, the various alternatives should not have a significant effect on the program. Beneficial uses of the sites by future generations would not be hindered.

6.1.8.6 Cost Analysis

Space disposal costs can be identified as follows (Bechtel 1979a):

- Waste processing/encapsulation (this may be incremental for comparisons with other alternatives)
- Ground transportation
- Launch facilities and space hardware (reusable and expendable)
- Launch operations and decommissioning
- Geologic disposal of residual nuclear wastes.

Although many of the basic space and waste technologies are understood, extrapolation to meet the requirements of the space disposal mission does not permit a valid cost estimate at this conceptual stage of the program. Initial scoping studies indicate that costs for many of these portions of the space disposal system would be similar to costs for other alternatives. The major cost difference for the space disposal alternative is attributable to the Space Shuttle operations. Capital, operating, and decommissioning costs for this incremental portion of the program are discussed briefly below.

Capital Costs

Capital costs would be incurred at Kennedy Space Center for construction of equipment dedicated to the waste disposal mission. This would include the special purpose transporter, launch pad, launch platform, and firing room. If these capital costs were recovered as charges to DOE as a Space Shuttle user, as is contemplated for other Space Shuttle applications, they would accrue as operating costs to any DOE space disposal program. Therefore, these costs would be integrated in the per-flight charges under operating costs. One special facility not usable for other shuttle operations would be the payload preparation facility. Current estimates for this facility are $29 million (1978 dollars). Other capital costs might accrue because of the need to allow radiation to decay in the HLW for at least 10 years prior to space disposal. Costs for such interim storage facilities have not been identified at this time.

Operating Costs

Operating costs for the space disposal alternative would be calculated on a per-flight basis, as they are for other participants in the Space Shuttle program. The per-flight cost would be approximately $39 million in 1978 dollars.

The breakout of this estimate is:

- Uprated Space Shuttle - $16 million
- Orbit transfer vehicle - $1.6 million
- Solar orbit insertion stage - $1.6 million
- Reentry vehicles -$5 million.

Decommissioning Costs

Decommissioning costs associated with Space Shuttle waste disposal operations would probably be limited to the facilities for waste processing and packaging, the only facilities at which contamination might be anticipated. Those decommissioning costs have been estimated at 10 percent of the initial capital costs, i.e., approximately $3 million. Costs for decommissioning other facilities associated with the space disposal alternative are assumed to be similar to those for decommissioning facilities associated with other waste disposal alternatives.

6.1.8.7 Safeguard Requirements

Safeguards would be considered for both space disposal and the associated terrestrial disposal. For space disposal of HLW, the risk of diversion would be short-term. Once the waste had been successfully disposed of in accordance with the design, the probability of an unauthorized retrieval would be very low. Physical protection of the sensitive facilities and transportation operations would be the most effective way to deny access for the short term. Note that if this alternative were chosen for the once-through fuel cycle, despite the very high throughput required, on a purely safeguards basis it would compare favorably with many other alternatives because of the difficulty of retrieving material once it is successfully deployed. See Section 4.10 for further details on safeguards for applicable predisposal operations.

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6.2 COMPARISON OF ALTERNATIVE WASTE DISPOSAL CONCEPTS

This section provides an assessment of the nine waste management concepts discussed in Chapter 5 and Section 6.1 of this Statement.

For the reader's convenience, a brief review of each of the alternative concepts is first presented in Section 6.2.1. Next, ten assessment factors and a set of related standards of judgement are introduced. The first stage of the analysis follows, in which the concepts are screened using the standards of judgement introduced in the previous section. Concepts which remain after the screening are then compared on the basis of the assessment factors and most promising concepts identified.

6.2.1 Summary Description of Alternative Waste Disposal Concepts

This section presents brief descriptions of the nine waste management concepts considered in this comparison. Characteristics of each concept are described in more detail in Chapters 4 and 5 and Section 6.1. Technical approaches not summarized here have been advanced for certain concepts that if implemented might result in a waste management system differing from that described here. In addition, the developmental process might result in a system different than described here, especially for concepts currently in a very preliminary stage of development.

6.2.1.1 Mined Repository

In the mined repository concept, disposal of waste would be achieved by manned emplacement in mined chambers in stable geologic formations. Engineered containment would be provided by the waste form, canisters, overpacks, and sleeves. Use of a tailored backfill would provide an additional engineered barrier. Isolation and natural barriers would be provided by the host rock and surrounding geologic environment, which would be selected to provide stability, minimal hydrologic transport potential and low resource attractiveness.

A waste packaging facility would be located at the repository site where spent fuel assemblies would be individually sealed into canisters. The canisters would be incorporated into the multibarrier package and then would be placed in individual boreholes in the floor and walls of mined chambers 500 to 1,000 m deep in suitable host-rock formations. Backfill would be placed around each package following emplacement. As each chamber is ready, it would be backfilled with rock and sealed. When the repository is filled the access tunnels and shafts would be filled with appropriate materials and sealed.

All waste types referenced in Table 6.2.1 could be emplaced in the mined repository.

A reprocessing fuel cycle would produce high-level liquid waste that could be solidified to a stable waste form, packaged in canisters that are part of a multibarrier package, and emplaced in the mined repository. Transuranic waste[a] would also be packaged and emplaced in the mined repository.

(a) Hulls, hardware, remotely handled and contact-handled TRU waste. See Table 6.2.1.

TABLE 6.2.1. Disposition of Principal Waste Products Using the Proposed Waste Disposal Concepts

	Spent Fuel Assemblies	High-Level Liquid (Fuel Processing Waste)	TRU Waste[a]
Mined Repository	Packaged and emplaced in mined repository.	Incorporated in immobilized solid, packaged and emplaced.	Packaged and emplaced in mined repository.
Very Deep Hole	Packaged and emplaced in deep hole repository.	Converted to immobilized solid. Packaged andemplaced in deep hole repository.	Disposal using suitable alternative technique.[b]
Rock Melt	Processed to a liquid state	Poured in rock melt repository.	Disposal using suitable alternative technique.[c]
Island Mined Repository	Packaged and emplaced in island mined repository.	Converted to immobile solid. Packaged and emplaced in island repository.	Packaged and emplaced in island mined repository.
Subseabed	Packaged and emplaced in subseabed repository.	Converted to immobile solid. Packaged and emplaced in subseabed repository.	Disposal using suitable alternative technique.[b]
Ice Sheet	Packaged and emplaced in ice sheet repository.	Converted to immobile solid. Packaged and emplaced in ice sheet repository.	Disposal using suitable alternative technique.[b]
Well Injection	Processed	Injected into geologic formations.	Disposed using suitable alternative concept.
Transmutation	Processed	Selected isotopes partitioned and transmuted to stable or shorter lived isotopes and disposed of using alternative concept.	Disposed using suitable alternative concept.
Space	Processed	Entire waste stream or selected isotopes converted to solid and emplaced in heliocentric orbit.	Disposed using suitable alternative concept.

(a) Remotely handled and contact-handled TRU wastes including dissolver solids, HEPA filters, incinerator ash wastes, failed and decommissioned equipment wastes.
(b) Could possibly be disposed of by the concept, but this is considered unlikely.
(c) Some chopped cladding and TRU wastes might be slurried into rock melt cavity subject to diluting limitations on HLW waste.

6.2.1.2 Very Deep Hole

In the very deep hole concept, disposal of high-level waste would be achieved by remote emplacement in bored shafts at depths greatly exceeding those of the mined repository. Engineered containment would be provided by the waste form, canisters, and perhaps additional barrier layers. Sorptive backfill, if used, would provide an additional engineered barrier. Isolation and natural barriers would be provided by the host rock and surrounding geologic and hydrologic environment, enhanced by the great distance to the accessible environment. The geologic and hydrologic environment would be selected to provide stability, minimal hydrologic transport potential, and low resource attractiveness.

A waste packaging facility would be located at the repository site where spent fuel assemblies would be packaged individually. The packaged fuel assemblies would be placed in rotary drilled holes as much as 10,000 m deep in crystalline rock. Holes for packages for fuel assemblies would be approximately 48 cm in diameter. After emplacement of approximately 150 packages in the bottom 1,500 m of the hole, the hole would be sealed and filled.

A reprocessing fuel cycle would require that prior to emplacement, high-level liquid waste be converted to an immobile solid and incorporated into a multibarrier package compatible with the very-deep hole environment. TRU waste resulting from reprocessing would be disposed using other suitable disposal concepts (Table 6.2.1).

6.2.1.3 Rock Melting

In the rock melting concept, disposal of high-level and some TRU waste would be achieved by remote emplacement of liquid or slurried waste into a mined cavity. Decay heat would be allowed to melt the surrounding rock which eventually would solidify, and form a solid, relatively insoluble, rock-waste matrix. Engineered containment could be provided during the operational period by a temporary chamber lining; however, engineered barriers would not be present during the molten phase. Following solidification, the rock-waste matrix would provide quasi-engineered containment wherein the host rock and waste forms would provide suitable post-solidification properties. Isolation and natural barriers would be provided by the surrounding geologic and hydrologic environment which would be selected to provide stability, minimal hydrologic transport potential and low resource attractiveness.

Spent fuel would be converted to a slurry or dissolved at a waste processing facility located at the repository site. Plutonium and uranium could be chemically separated and sent to a mixed oxide fuel fabrication facility if a reprocessing fuel cycle were utilized. High-level waste and contact-handled TRU waste in liquid or slurry form would be piped separately to the repository. Here the waste would be injected into mined cavities approximately 20 m in diameter and 2,000 m deep. Liquid or slurried contact-handled TRU waste, supplemented with water as required, could be injected into the cavity to provide cooling. After the cavity is filled, cooling would be terminated and the injection shaft sealed. Heat from radioactive decay would melt the surrounding rock, forming a molten rock-waste mix at a temperature $\gtrsim$1000°C. The mix would eventually solidify, trapping the waste within a rock matrix. Solidification should be complete in about 1,000 years.

Fuel hardware and TRU waste for which conversion to liquid or slurry is impractical would be packaged and emplaced using a suitable alternative disposal concept (Table 6.2.1).

6.2.1.4 Island Mined Repository

In the island mined repository concept, disposal of waste would be achieved by manned emplacement in mined chambers in stable geologic formations on continental islands. Engineered containment would be provided by the waste form and multibarrier package. Tailored sorptive backfill would provide an additional engineered barrier. Isolation and natural barriers would be provided by the host rock and the surrounding geologic and hydrologic environment which would be selected to provide stability, minimal hydrologic transport potential and low resource attractiveness.

Spent fuel assemblies would be packaged individually into canisters at a waste packaging facility located in the continental U.S. All canisters would be loaded into shipping casks and transported by rail to the embarkation port facility. At the port facility the waste packages would be transferred from the rail casks to ocean shipping casks which would be loaded aboard ocean-going vessels. These vessels would transport the waste to a receiving port on the U.S.-owned repository island. Waste casks would be transferred to rail or highway vehicles for shipment to the repository site. Here the canisters would be unloaded from the shipping casks, placed in multibarrier packages, and placed in individual boreholes in the floor of mined chambers at least 500 m deep in granite or basalt, located either within the fresh groundwater lens or within underlying saline groundwater. Backfill would be placed around each package following emplacement. As each chamber is ready it would be backfilled and sealed. When the repository is filled the access tunnels and shafts would be backfilled with appropriate materials and sealed.

A reprocessing fuel cycle would require high-level liquid waste to be converted into an immobile solid that would be incorporated into a multibarrier package compatible with the island geologic environment. Other wastes would be packaged and emplaced in the island repository.

6.2.1.5 Subseabed Disposal

In the subseabed disposal concept, disposal of waste would be achieved by remote emplacement in relatively thick, stable beds of sediment located in deep, quiescent, and remote regions of the oceans. Engineered multibarrier containment would be provided by the waste form, canister, and the outer body of the emplacement container. Isolation and a natural barrier would be provided by clay sediments which would be chosen for uniformity, high plasticity, low permeability, high sorption potential, long-term stability and low resource attractiveness. The ocean itself would enhance remoteness, providing protection from human intrusion. Because the ocean is part of the accessible environment it would not be considered as a barrier to waste release.

Spent fuel assemblies would be packaged individually in canisters at a waste packaging facility located in the continental U.S. Packaged fuel assemblies would be loaded into shipping casks and transported by rail to the embarkation port facility. At the port facility waste packages would be removed from the shipping casks and loaded into emplacement vehicles, probably free fall penetrometers. These would be loaded onto special oceangoing vessels and transported to the emplacement site, located in the mid-plate, mid-gyre region of the ocean with depths of 3,000 to 5,000 m. At the site the penetrometers would be released to penetrate 50 to 100 m into the clay sediment. Closing of the hole above the penetrometers might occur spontaneously or be accomplished by mechanical means and would seal the waste into the sediment. A monitoring vessel would verify satisfactory emplacement.

A reprocessing fuel cycle would produce liquid high-level waste that would be converted to an immobile solid for incorporation into a multibarrier package designed for emplacement in the sediments. TRU waste would probably require another suitable disposal concept (Table 6.2.1).

6.2.1.6 Ice Sheet Disposal

In the ice sheet disposal concept, disposal of high-level waste would be achieved by remote emplacement within a continental ice sheet. The plasticity of the ice would eventually seal the waste from the environment and subfreezing temperatures would preclude hydrologic transport except possibly at the conditions encountered at the ice-rock interface. Engineered multibarrier containment would be provided by the waste form and canisters and possibly overpacks. Isolation and a natural barrier would be provided by the ice mass. The geographic location of the repository and the inclement weather of continental ice sheets would contribute to the remoteness of the repository and decrease the possibility of human intrusion.

Spent fuel assemblies would be packaged individually in canisters at a waste processing facility located in the continental U.S. Packaged fuel assemblies would be loaded into shipping casks and transported by rail to the embarkation port facility. At the port facility waste packages would be transferred from rail casks to ocean-shipping casks which would be loaded aboard ocean-going vessels. These vessels would transport the waste to a receiving port at the ice margin. Here the waste packages in shipping casks, would be transferred to tracked vehicles for transport to the repository, located some distance inland. At the repository site the waste packages would be removed from the transport casks, placed into pilot holes drilled 50 to 100 m into the ice and tethered to anchor plates with 200 to 500 m cables or allowed to melt freely into the ice. Heat from radioactive decay would melt the ice and the package would sink into the ice sheet, reaching its final position in six to eighteen months. The pilot holes would be sealed by filling with water which would subsequently freeze. Refreezing of water above the package as it progressed downward would complete sealing of the emplacement holes.

A reprocessing fuel cycle would produce liquid high-level waste that would be converted to an immobile solid compatible with the ice environment. This solidified waste would be packaged and emplaced in the ice sheet repository. TRU waste would probably be disposed using an alternative disposal concept (Table 6.2.1).

6.2.1.7 Well Injection

In the well injection disposal concept, disposal of high-level waste would be achieved by remote emplacement of liquid or slurried waste into stable geologic formations capped by an impermeable boundary layer. A degree of engineered containment would be supplied by the waste form if a grout were used but would not be present during the injection phase. Isolation and natural barriers would be provided by the host rock and the surrounding geologic and hydrologic environment which would be selected for its stability, minimum hydrologic transport potential, high sorption potential and low resource attractiveness.

A waste processing facility would be located at the repository site where spent fuel would be dissolved and prepared for injection, either directly as a dilute acidic liquid or as a neutralized grout. The prepared waste would be transferred by piping to the injection well field. Dilute acid waste, if used, would be injected into porous sandstone having shale caprock at depths of approximately 1,000 m. Neutralized grout would be injected into a shale formation having natural or induced fractures at depths of approximately 500 m. TRU waste would require an alternative disposal concept.

Liquid high-level waste resulting from a reprocessing fuel cycle would be transferred directly to the waste preparation facility, colocated with the reprocessing plant. TRU waste would be packaged and emplaced using an alternative disposal concept (Table 6.2.1).

6.2.1.8 Transmutation

Transmutation would function as an ancillary waste treatment process for the conversion of selected long-lived waste isotopes to shorter-lived isotopes potentially reducing the time during which repository integrity must be maintained. The process would be operated in conjunction with a waste management system using a suitable alternative disposal concept for disposal of radioactive waste, including transmutation products (Table 6.2.1). Because transmutation is a waste treatment process and not a disposal alternative, it cannot be assessed in terms of containment, barriers and remoteness in the same manner as these terms are applied to repositories.

At a processing plant spent fuel would be dissolved and uranium and plutonium separated for recycle. Reprocessing wastes would be transferred to an adjacent partitioning facility where long-lived waste isotopes would be partitioned from the reprocessing waste stream. The residual waste streams, stripped of long-lived isotopes, would be processed for disposal using a suitable disposal concept. The isotopes selected for transmutation would be combined with recovered plutonium and uranium and shipped to a MOX-FFP.

At the fuel fabrication plant the plutonium-uranium-waste isotope mixture would be fabricated into MOX fuel assemblies following addition of sufficient enriched uranium to achieve the desired end-of-cycle reactivity. TRU waste from the fuel fabrication plant would be sent to a colocated waste purification facility for recovery of waste actinides. Recovered actinides would be returned to the fuel fabrication facility for incorporation into MOX fuel; the residual waste would be processed for disposal using a suitable alternative waste disposal concept (Table 6.2.1).

The MOX fuel, containing the waste isotopes for transmutation, would be shipped in shielded casks to power reactors where a portion of the waste isotopes would be transmuted to stable or shorter-lived isotopes. Transmuted isotopes would be partitioned for disposal during the subsequent reprocessing cycle. Repeated recycles would be required to achieve nearly complete transmutation of the long-lived isotopes.

Implementation of transmutation as an actinide waste treatment process requires that spent fuel be reprocessed to recover the actinides and that the actinides be recycled for transmutation, mandating a reprocessing-type fuel cycle.

6.2.1.9 Space

In the space disposal concept, disposal of selected waste products would be achieved by insertion of waste packages into a stable solar orbit approximately half-way between the orbits of Earth and Venus. Engineered containment would be provided by the waste form and its engineered package. Isolation would be provided by the remoteness of the orbit from Earth and the stability of the orbit. An additional impediment to return of waste would be provided by inclining the orbit to the ecliptic.

Spent fuel would be chopped and dissolved at a processing facility. Plutonium and uranium would be chemically separated and sent to a MOX-FFP if a reprocessing fuel cycle were utilized. Waste products for which space disposal is intended would be partitioned from the waste stream and transferred to an adjacent waste preparation facility. High-level and contact-handled TRU waste not destined for space disposal would be processed for disposal using a suitable alternative disposal concept (Table 6.2.1). Alternatively, the entire liquid high-level waste stream, including uranium and plutonium constituents, could be transferred to the waste preparation facility for space disposal.

At the waste preparation facility, the waste would be incorporated into a solid ceramic-metal composite ("cermet") which would be formed into a payload of suitable shape and size. The payload would be packed into a radiation shield and this assembly loaded into a shipping cask for transport to the nuclear payload preparation facility near the launch site.

At the nuclear payload preparation facility, the shielded waste assembly would be removed from the shipping cask and loaded into a reentry vehicle. A special transporter would then take the assemby to the launch site, where it would be positioned in the space shuttle cargo bay with an orbit transfer vehicle and a solar orbit insertion stage.

The space shuttle would be launched into earth orbit where the reentry vehicle-payload assembly would be deployed from the cargo bay. The shielded waste assembly would then be removed from the reentry vehicle and attached to the solar orbit insertion stage, which would be latched to the orbit transfer vehicle. The orbit transfer vehicle would propel the solar orbit insertion stage into an earth escape trajectory, release the solar orbit insertion stage and return to earth orbit for recovery. The solar orbit insertion stage and the waste would continue and the waste would ultimately be inserted into a stable solar orbit at 0.85 astronomical units. The space shuttle would return to earth carrying the reentry and orbit transfer vehicles.

6.2.1.10 Summary

The relationships of the nine disposal concepts to the waste products of the two primary fuel cycles have been summarized in Table 6.2.1. Products of the once-through fuel cycle include spent fuel assemblies with probably a small stream of contact-handled TRU waste resulting from fuel element failures. Five of the disposal concepts could dispose of these products directly. However, rock melt, well injection, transmutation and space disposal would require processing the spent fuel to liquid or slurry form with the result that the spectrum of waste products characteristic of the reprocessing fuel cycle is generated. This includes liquid high-level waste, fuel hulls and hardware, and a substantial quantity of remotely handled and contact-handled TRU waste. It should be noted that the reprocessing fuel cycle will likely require an alternative disposal facility (probably a mined repository) for the high volume TRU wastes for all concepts except the island repository; mined repositories; and, perhaps, the subseabed.

6.2.2 Assessment Factors and Standards of Judgement

Ten assessment factors have been selected to facilitate comparison of the proposed waste management concepts. These factors are discussed in Subsections 6.2.2.1 through 6.2.2.10. Associated with certain of these factors are standards of judgement. The standards of judgement are applied in Section 6.2.3 to reduce the nine proposed waste management concepts to a subset of candidate concepts with greatest potential for adequate performance. Concepts in this subset are then compared in Section 6.2.4 on the basis of the ten assessment factors. The ten assessment factors are listed in Table 6.2.2 below; the assessment factors are underlined. The standards of judgement appear as bullets in Table 6.2.3 and are grouped under the (underlined) assessment factors.

TABLE 6.2.2. Assessment Factors

Radiological Effects
- operational period
- post-operational period

Non-Radiological Environmental Effects
- health effects
- socio-economic effects
- aesthetic effects
- ecosystem effects

Current Status of Development
- availability of technology
- availability of performance assessment methodologies

Conformance with Federal Law and International Agreements

Independence from Future Development of the Nuclear Industry
- industry size
- fuel cycles
- reactor design

Cost of Development and Operation

Potential for Corrective or Mitigating Action

Long-Term Maintenance and Surveillance Requirements

Resource Consumption

Equity of Risk

TABLE 6.2.3. Standards of Judgement

Radiological Effects

- A concept should comply with radiological standards established for other fuel cycle facilities.
- Containment should be maintained during the period dominated by fission product decay.
- Waste should be isolated from the accessible environment for a minimum of 10,000 years.

Non-Radiological Environmental Effects

No standards were advanced for this factor.

Current Status of Development

- The concept should be amenable to development within a reasonable period of time such that implementation is not left to future generations.
- Implementation of a concept should not require scientific breakthroughs.
- Capabilities for assessing the performance of a concept must be available prior to committing major R&D programs to its development.

Conformance with Federal Law and International Agreements

No standards were advanced for this factor.

Independence from Future Development of the Nuclear Industry

- Implementation of a concept should not be dependent upon the size of the nuclear industry.
- Concepts should be independent of fuel cycle issues.
- Concepts should be independent of reactor design issues.

Cost of Development and Operation

No standards were advanced for this factor.

Potential for Corrective or Mitigating Action

- Concepts should allow corrective action to be taken in case of failure of a system to perform as designed.

Long-Term Maintenance and Surveillance Requirements

- Reliance should not be placed on maintenance or surveillance for extended times following termination of the operational period.

Resource Consumption

No standards were advanced for this factor.

Equity of Risk

No standards were advanced for this factor.

6.2.2.1 Radiological Effects

A central objective of the nuclear waste management program is to limit radiation dose to both the public and to operating personnel to acceptably low levels. Two time periods are of interest. One is the operational period involving waste treatment, transportation, and emplacement and the second is the post operational period following termination of repository operations.

A useful measure of radiological effects during the operational period is radiation exposure resulting from emplacement of a quantity of waste derived from the generation of a unit of electrical power by nuclear means. Unfortunately, the current state of development of many of the concepts does not permit computation of this measure. Therefore, this analysis will rely upon relative comparison, using processing and transportation requirements as secondary indicators of potential radiation dose during the operational period.

A reasonable minimum level of radiological performance during the operating period is that risks shall not be greater than those allowed for other nuclear fuel cycle facilities. This suggests a standard that appropriate regulatory requirements established for other fuel cycle facilities be met.

Objectives 1 and 2 of the proposed DOE Waste Management Performance Objectives (Table 6.2.4) are intended to provide standards related to the radiological performance of waste management concepts during the post-emplacement period. Objective 1 requires that waste containment within the immediate vicinity of initial placement should be virtually complete during the period when radiation and thermal output are dominated by fission product decay. Objective 2 requires a standard of reasonable assurance that wastes will be isolated from the environment for a period of at least 10,000 years with no prediction of significant decrease beyond that time. Both standards were adopted for this analysis (Table 6.2.3).

TABLE 6.2.4. Proposed DOE Waste Management Performance Objectives[(a)]

1. Waste containment within the immediate vicinity of initial placement should be virtually complete during the period when radiation and thermal output are dominated by fission product decay. Any loss of containment should be a gradual process which results in very small fractional waste inventory release rates extending over very long release times, i.e., catastrophic losses of containment should not occur.
2. Disposal systems should provide reasonable assurance that wastes will be isolated from the accessible environment for a period of at least 10,000 years with no prediction of significant decreases in isolation beyond that time.
3. Risks during the operating phase of waste disposal systems should not be greater than those allowed for other nuclear fuel cycle facilities. Appropriate regulatory requirements established for other fuel cycle facilities of a like nature should be met.
4. The environmental impacts associated with waste disposal systems should be mitigated to the extent reasonably achievable.
5. The waste disposal system design and the analytical methods used to develop and demonstrate system effectiveness should be sufficiently conservative to compensate for residual design, operational, and long-term predictive uncertainties of potential importance to system effectiveness, and should provide reasonable assurance that regulatory standards will be met.
6. Waste disposal systems selected for implementation should be based upon a level of technology that can be implemented within a reasonable period of time, should not depend upon scientific breakthroughs, should be able to be assessed with current capabilities, and should not require active maintenance or surveillance for unreasonable times into the future.
7. Waste disposal concepts selected for implementation should be independent of the size of the nuclear industry and of the resolution of specific fuel cycle or reactor design issues and should be compatible with national policies.

(a) DOE/NE-0007--Statement of Position of the United States Department of Energy in the Matter of Proposed Rulemaking on the Storage and Disposal of Nuclear Waste.

Non-Radiological Environmental Effects

Non-radiological environmental effects considered to be of potential significance in the comparison of waste management concepts include health effects from non-radiological causes, socioeconomic effects, aesthetic effects, and effects on ecosystems.

Health effects from non-radiological causes include injuries and deaths occurring to both occupational workers and to the general public from routine operations and from accidental conditions.

Socioeconomic effects include impacts on the well-being of communities in the vicinity of waste management facilities.

Potential aesthetic effects include noise, odor and impacts on visual resources.

Both natural and managed ecosystems would be affected by waste management operations. Potential impacts include those on ecosystem productivity, stability, and diversity.

No standards of judgement have been advanced for non-radiological environmental effects, although all concepts would be expected to comply with standards established by responsible Federal and state regulatory agencies. The proposed DOE Performance Objective 4 asserts the importance of minimizing non-radiation-related environmental effects.

6.2.2.2 Status of Development

This factor is intended to assess the waste management concepts on the basis of the maturity of the concepts. Two issues are of concern: 1) availability of technology required to implement the concept, including that required for site characterization, repository development, waste treatment, handling, emplacement, and monitoring; and, 2) ability to predict performance of the waste management system. A third issue, cost of research and development, is considered under the factor of cost.

Three standards of judgement relating to status of development can be derived from the proposed DOE Performance Objective 6. First the technology must be implemented within a reasonable period of time where "reasonable period of time" implies that those currently responsible can complete the major part of implementing a concept and not pass an unresolved problem on to future generations. Consequently, Objective 6 also states that scientific breakthroughs should not be required to permit implementation of a concept. Further capabilities for assessing the performance of any particular waste management concept must be available at the time that a decision is made to place emphasis on the development of any particular concept.

6.2.2.3 Conformance with Federal Law and International Agreements

The purpose of this factor is to identify and compare potential conflicts with Federal legislation and international treaties, conventions, and understandings to which this nation is a party that would prevent implementation of a proposed option. The DOE proposed Performance Objective 7 states that waste management systems "should be compatible with national policies" suggesting that concepts might be rejected because of potential policy conflicts. Because Federal legislation and international agreements can be amended for reasonable cause, this condition will not be used as a standard, but its consideration provides insight into the difficulty of implementation. Any waste management concept, if implemented, would be required to comply with applicable laws and regulations.

6.2.2.4 Independence from Future Development of the Nuclear Industry

Implementing a nuclear waste management system is a large scale, costly, and long-term effort. Concepts selected for priority development should be independent of the future development of the nuclear industry including industry size, fuel cycles, and reactor designs.

Three standards of judgement derived from DOE Performance Objective 7 are related to this factor: 1) waste disposal concepts selected for implementation should be independent of the size of the nuclear industry, 2) independent of specific fuel cycles and 3) independent of reactor design issues.

6.2.2.5 Cost of Development and Operation

The purpose of this factor is to compare concepts on the basis of estimated costs for research and development (presumably to be borne by the Federal government but recovered from the utilities through fees charged for disposal) and on costs of implementation and operation (borne by utilities and included in their rate bases). No standards have been established for cost.

6.2.2.6 Potential for Corrective Action

The probability of system failure can be reduced to low levels by careful design, thorough assessment of performance and provision of redundant systems. However, as with any engineered system, probability of failure cannot be entirely eliminated, with the result that there will remain a probability (although very low) that the system may not perform as expected. Thus the ability to detect and correct failure or to mitigate its consequences would be a desirable property of the concept selected for implementation. The desirability of corrective action capability is implied by DOE Performance Objective 5 which suggests that corrective action capabilities should be provided to compensate for residual uncertainties in system performance. Thus the importance of corrective action capability should be assessed with consideration of residual uncertainties in system performance.

The proposed NRC Technical Standards for Regulating Geologic Disposal of High-Level Radioactive Waste require retrievability, a form of corrective action, to be maintained for 50 years following termination of waste emplacement operations (Proposed 10 CFR 60.111(a)(3)). No standards were established for corrective action potential given the dissimilar characteristics of certain of the waste management options.

6.2.2.7 Long-Term Maintenance and Surveillance Requirements

Future generations cannot reasonably be expected to assume a burden of maintaining and monitoring the nuclear wastes of present generations. Thus a desirable assessment factor for waste management concepts is that they require minimal maintenance or monitoring following decommissioning. The Environmental Protection Agency has included in its draft standards for waste management a stipulation that surveillance and maintenance should not be relied upon for a period exceeding 100 years after termination of active disposal operations (43 Fed. Register, Section 221, November 1978). A more general performance standard was adopted for this analysis that reliance should not be placed on maintenance and surveillance for extended times following termination of the operational period.

6.2.2.8 Resource Consumption

Any waste management option would require the consumption of certain resources including energy, critical nonfuel materials, and land. Certain materials which are important to a waste management option may be in short supply, potentially producing market disruptions or increased dependence on uncertain supplies. Potentially critical materials are listed in Table 6.2.5. It is important that no waste isolation approach use an unreasonable amount of any critical resource, but no specific standard is advanced.

TABLE 6.2.5. Potentially Critical Materials[(a)]

Aluminum	Cobalt	Nickel	Water
Antimony	Columbium	Platinum	Natural Gas
Asbestos	Graphite	Potash	Electricity
Bismuth	Iodine	Quartz (crystals)	Coal
Cesium	Manganese	Tantalum	Petroleum-Derived Fuels
Chromium	Mica	Tin	Other Fuels

(a) The nonfuel minerals of this group are considered to be "major problems from the national viewpoint" by the U.S. Bureau of Mines because of U.S. low-grade resource or reserve inadequacy to Year 2000

6.2.2.9 Equity of Risk

Although the responsibility for disposal of high level radioactive waste belongs to the Federal government, the implementation of a specific solution will require cooperation with the state and local governments, and with the general public. A few localities will be required to accept and service the facilities for disposal of waste that was created in providing service and benefits to a very broad segment of the country's population. Consequently, the implementation of a disposal method will have to be judged against the equity of risk by the political subdivision involved.

6.2.3 Application of Performance Standards

The nine proposed waste disposal concepts are examined in this section with respect to the performance standards advanced in Table 6.2.3. Results of this judgement are tabulated in Table 6.2.6. The subset of concepts meeting these standards are subjected to more detailed comparative analysis in Section 6.2.4.

6.2.3.1 A Concept Should Comply with Radiological Standards Established for Other Fuel Cycle Facilities

The unique characteristics of several of the proposed waste disposal concepts set them quite apart in design and operation from any existing fuel cycle facility. Thus, although it is appropriate to evaluate the concepts on current dose, risk and emission standards, it may be inappropriate to apply regulations relating to the means of achieving these standards. It is not evident, based on available information, that any of the nine proposed concepts would necessarily fail to comply with dose, risk and emission standards; though it is likely that the radiological releases would vary among the concepts.

6.2.3.2 Containment Should be Maintained During the Period Dominated by Fission Product Decay

"Containment" is defined in the NRC proposed technical criteria for regulating geologic disposal of high-level radioactive waste as "keeping radioactive waste within a designated boundary" (Proposed 10 CFR Part 60). Because of inherent differences among the concepts, the following definitions of containment are used for this assessment:

- Mined Repository--Waste is contained within the waste package (Proposed 10 CFR Part 60.)
- Very Deep Hole, Island Mined Repository, Ice Sheet Disposal: Waste is contained within the package.
- Rock Melt--Waste is contained within the rock-waste matrix, and in the intended location.
- Subseabed Disposal--Waste is contained within the package (penetrometer case or overpack).
- Well Injection--Dilute Acid: Waste is contained within the intended region of the host formation
  Shale-Grout: Waste is contained within the grout matrix, and in the intended region of the host formation.
- Transmutation--None, the containment concept is not applicable.
- Space--Waste is contained within its package within the predetermined heliocentric orbit.

Based on these definitions of containment, engineering judgement indicates that containment for several hundred years could likely be achieved using the mined repository, very deep hole, island mined repository, subseabed, ice sheet, and space disposal concepts. Uncertainties, however, are associated with the very deep hole concept depending on depth of emplacement and associated conditions of temperature and pressure to which the package is exposed.

Because the rock melt concept does not provide a system of engineered barriers, and because of the elevated temperatures, it appears likely that heated water vapor or liquid could contact, leach and transport waste from the as yet unsolidified rock-waste matrix of the rock melt concept during the initial 1000-year post-operational period.

Because the well injection concept does not provide a series of engineered barriers, one thousand year containment could not be assured with either of the well injection proposals. Diffusion of dilute acid injected waste into fractures and discontinuities of formations adjacent to the host formation could be expected.

In conclusion, it appears probable that containment of emplaced waste, as defined, could be maintained through the period dominated by fission product decay for all concepts except rock melt and well injection. The containment concept does not apply to transmutation.

6.2.3.3 Waste Should Be Isolated from the Accessible Environment for a Minimum of 10,000 Years

Ten thousand years has been proposed as a time period during which the radiotoxicity of properly treated waste would decay to levels comparable with the natural uranium ore bodies from which the materials were originally derived (Voss 1980). "Isolated" is interpreted as "segregation of the waste from the accessible environment within acceptable limits" (Proposed 10 CFR Part 60) where the accessible environment includes the atmosphere, the land surface, surface waters, oceans and presently used aquifers (Proposed, 10 CFR Part 60, 40 CFR Part 146). "Acceptable limits" has been generally interpreted to include releases resulting in dose rates within the normal variation of naturally occurring radiation dose rates (DOE 1980).

Analysis to date of the mined repository concept suggests no reason to believe that acceptable isolation could not be maintained by the geologic environment for a 10,000-year period, with the possible exception of very low probability catastrophic accident situations. The probability of these occurring is estimated to be small. Similarly, it appears quite possible that the very deep hole concept could maintain acceptable waste isolation over the required period if such depths are successfully isolated from ground water.

Maintenance of waste package containment cannot be assumed for the 10,000-year period for the mined repository, very deep hole, island mined repository, subseabed disposal and ice sheet disposal concepts. Package failure would expose the waste form to a saturated hydrologic environment for the subseabed and island disposal concepts and acceptable isolation would be dependent upon stability of the hydrologic environment and the sorptive properties of the host material and surrounding geologic environment. Available evidence indicates that acceptable isolation could be maintained using the subseabed concept. Satisfactory performance of the island concept, while possible, is less certain because of an incomplete understanding of island hydrologic systems.

Maintenance of isolation for the requisite period under ice sheet conditions appears to be sufficiently questionable as to preclude this option from further consideration on the basis of this standard of judgement. If not tethered, the packages would descend to the ice-rock interface where the waste form packages could be pulverized by ice motion, and waste subsequently transported to the ocean by water potentially present at the interface. If tethered, ice sheet erosion or sublimation (possible within a 10,000-year period given historical climatic fluctuation) could expose waste to the surface environment.

The waste-rock matrix of the rock melt concept would potentially be exposed to severe hydrothermal alteration and leaching conditions late in the cooling phase when hot water may be present at the periphery of the rock-waste mass. This could result in transfer of waste to ground water. However if the surrounding geologic and hydrologic conditions were suitable, migration of waste to the accessible environment might be limited to acceptably low levels. On the other hand, thermomechanical disruption of the surrounding geology by the rock melt process might allow rapid transfer of contaminated ground water to surface aquifers, especially if promoted by thermal gradients from decay heat. While there is currently insufficient evidence to eliminate rock melt from further consideration on the basis of this standard of judgement, satisfactory performance appears highly uncertain. Furthermore a method for resolving this uncertainty does not appear to be available.

The host rock is the primary isolation mechanism for the shale-grout version of well injection. Assuming a suitably stable formation of adequate sorptive potential, preliminary calculations (Section 6.1.6) indicate that the likelihood of unacceptable quantities of radionuclides reaching accessible ground water is small. For dilute acid injection, assuming the site has suitable bounding formations, it also appears that there would be a low probability of unacceptable quantities of radioisotopes reaching accessible aquifers. However, prediction of acceptable long-term performance of well injection will require thorough characterization and understanding of the host formations and surrounding geology. It is highly uncertain at this time how this could be accomplished.

The transmutation concept may not require repositories providing 10,000-year isolation if all long-lived isotopes are eliminated. However, the 10,000-year isolation standard is not applicable to the transmutation process per se.

The space disposal concept appears to have most merit with respect to isolation. It has been calculated that a stable orbit would provide a minimum of 1 million years isolation.

In conclusion, it appears that all concepts with the exception of ice sheet, rock melt, and well injection have the potential of meeting the 10,000-year standard for acceptable waste isolation.

6.2.3.4 The Concept Should be Amenable to Development Within a Reasonable Period of Time Such That Implementation is Not Left to Future Generations

Necessary implementation time[a] for the ice sheet concept is estimated to be 30 years or greater (Section 6.1.5) primarily because of the substantial uncertainties which remain to be resolved regarding ice sheet stability, structure, and dynamics and understanding of waste-ice interaction. A minimum time of 20 years is also projected for transmutation (Section 6.1.7); it is unlikely that this concept could be implemented prior to the turn of the century given the need to resolve theoretical uncertainties, and establish siting criteria; and the time required for pilot plant development, construction, and testing, and construction of commercial-scale facilities.

Development time has not been projected for the well injection concept. Although the engineering requirements for this concept do not appear difficult, requirements for improved site characterization techniques, performance assessment methods and monitoring technology appear to be formidable. However it may be possible to implement this concept within 20 years.

The remaining 20 years of this century would appear to be adequate for implementation of any of the remaining concepts, if it is assumed that very deep holes may be less than 10,000 m deep.

In summary, it appears that all concepts with the exception of ice sheet and transmutation qualify on this standard of judgement.

6.2.3.5 Implementation of a Concept Should Not Require Scientific Breakthroughs

Several concepts would require significant extension of existing technology to achieve satisfactory implementation; but none of the concepts appear to require scientific breakthroughs. Transmutation might be most efficiently accomplished in a fusion reactor, which would require a scientific breakthrough.

6.2.3.6 Capabilities for Assessing the Performance of a Concept Must Be Available Prior to Committing Major R&D Programs to Its Development

The need for substantial additional performance assessment capabilities appears to exist for all concepts. While the mined repository will require refinement of performance assessment capabilities, it is believed that this will be achieved in the near future. Manned inspection of the emplacement location is currently being proposed by the NRC. If this should be applied to all concepts, it would eliminate subseabed, very deep hole, ice sheet, well injection, space, and probably rock melt concepts.

All concepts, with the exception of transmutation, space, and subseabed require further development of remote sensing capability for assessment of the characteristics of the potential host media. In addition, the well injection and rock melt concepts would require

(a) All estimates of time assume that the concept discussed receives priority for funding.

development of methods for prediction and measurement of waste location and configuration. The lack of predictive methods for the ice sheet concept appears sufficiently intractable at this time to preclude consideration of this concept.

6.2.3.7 Implementation of a Concept Should Not Be Dependent Upon the Size of the Nuclear Industry

The rock melt, transmutation and space options appear to be potentially sensitive to the size of the nuclear industry. The reference rock melting concept would require sufficient waste product to operate at least one cavity (40,000 MTHM equivalent waste) and succeeding increments would be equally as large. The minimum size of a rock melt cavity has not been determined, however, and it is possible that smaller increments would be feasible. Transmutation would require operating reactors for the transmutation step and a sufficiently large industry to justify the investment in specialized support facilities. Space disposal, as well, would require a sizable investment in specialized hardware, needing a substantial nuclear industry to justify this investment. This, however, is an economic question and does not intrinsically disqualify space disposal from consideration.

6.2.3.8 Concepts Should Be Independent of Fuel Cycle Issues

Fuel cycles treated in this document include the once-through cycle and full uranium-plutonium recycle; however other cycles are possible. Although the uranium-only fuel cycle was discussed in the draft of this Statement, review comments indicate that this cycle is not considered reasonable by the industry or the scientific community and therefore this cycle is not considered further. Additional fuel cycle issues relate to timing of fuel cycle implementation and defense wastes.

Once-Through and Reprocessing Fuel Cycles

As summarized in Table 6.2.1, the mined repository and island mined repository concepts would be capable of accommodating all waste products of both the once-through and reprocessing fuel cycles. Various considerations suggest the use of mined repositories for bulky equipment and for the considerable volume of TRU wastes, hulls, and hardware generated by the reprocessing fuel cycle for disposal concepts that cannot accommodate these wastes.

The rock melt and well injection options could find application with either the once-through or the reprocessing fuel cycles. Fuel processing would be required for the once-through cycle.

The space disposal concept, as well, could find application to either fuel cycle, however, partitioning of the waste as well as processing of spent fuel would be required if the once-through fuel cycle were used.

Transmutation would find its most promising application with the reprocessing fuel cycle. Processing and partitioning of spent fuel and recycle in a reactor would be required and alternative disposal technology would be needed for disposal of other transmutation waste products, high-level liquid fission product waste and fuel hulls and hardware.

Timing

The timing of implementation of a waste management system could potentially affect the feasibility of the concepts because of declining decay heat generation rates or by the availability of facilities required to implement the concept. Substantial reduction of decay heat rates prior to emplacement of spent fuel or high-level waste could conceivably affect the operation of the rock melt and the ice sheet concepts; however reduction in decay heat rates over the time frames being considered for deferred fuel cycles do not appear to be great enough to materially affect operation of either of these concepts. Postponement of waste disposal operations beyond the period when light water power reactors were the dominant commercial type could impact the transmutation concept by requiring alternative transmutation devices. However, alternative devices, including fast breeder fission reactors and fusion devices, may be available and probably superior to light water reactors (Croff et al. 1980). Thus it is not felt that any concept can be dismissed on the basis of timing alone.

Summary of Fuel Cycle Issues

In summary, it appears that all of the concepts offer some potential benefit with any fuel cycle and that none should be dismissed because of sensitivity to fuel cycle issues (although the case for transmutation with a once-through fuel cycle appears to be quite marginal). Pursuit of the rock melt, well injection, transmutation or space disposal concepts with either fuel cycle would require concurrent development of one of the concepts capable of disposing of TRU waste, probably a mined repository.

6.2.3.9 Concepts Should Be Independent of Reactor Design Issues

None of the concepts appear to be especially sensitive to reactor design issues.

6.2.3.10 Implementation of a Concept Should Allow Ability to Correct or Mitigate Failure

This standard tends to favor those concepts in which wastes may be readily retrieved if observations of their actual behavior under full-scale implementation reveal previously unanticipated defects in the disposal system. Mined geologic disposal lends itself most readily to this requirement although obviously attempts at transmutation could easily be abandoned if large-scale operations failed to work.

Those concepts in which retrieval from a large-scale system would be difficult or impossible fail to meet this requirement. These concepts include space disposal, rock melt, well injection, and under certain circumstances, ice sheet disposal.

6.2.3.11 Maintenance or Surveillance Should Not Be Required for Extended Periods Following Termination of Active Repository Operations

The resolidification period of 1,000 years required of the rock melt concept would appear to require surveillance for a substantial period to verify long-term stability and satisfactory containment of the molten mass. This is seen as sufficiently contrary to this standard of judgement as to prohibit preferred consideration of the rock melt option. The other concepts appear not to be affected by this consideration.

6.2.3.12 Summary

The performance of the nine proposed disposal concepts against the standards of judgement is summarized in Table 6.2.6. It should be emphasized that these conclusions are based largely on judgement of the authors, based in many cases on fragmentary or qualitative information. Of the nine proposed concepts, mined repository, very deep hole, island mined repository, subseabed, and space disposal have the potential for meeting all of the standards. A comparison of these five concepts is given in the next section.

6.2.4 Comparison of the Waste Disposal Concepts with Most Potential

This section compares the mined repository, island mined repository, very deep hole, subseabed and space disposal concepts on the basis of the assessment factors introduced in Section 6.2.2.

6.2.4.1 Radiological Effects

Operational Period

During the operational period, occupational exposure due to waste management would be dominated by that associated with waste processing. Transportation of TRU waste represents the greatest source of dose to the general public because of the large volume of material. Additional dose to both occupational workers and to the general public could result from accidents.

Occupational radiological effects attributable to processing operations would likely be quite similar for the mined repository, very deep hole, island mined repository, and subseabed options because the waste treatments are similar. Slightly greater occupational exposure could be expected with the very deep hole and subseabed options should it be decided to section bulky TRU-contaminated equipment for disposal by these options--an unlikely decision. Space disposal would require dissolution of spent fuel for both once-through and reprocessing fuel cycles, potentially resulting in greater radiological effects compared to the other options.

Transportation and handling requirements of spent fuel from power reactors to the waste treatment/packaging facilities would be approximately equivalent for each of the disposal concepts. The mined repository and very deep hole emplacement facilities could be colocated with the treatement/packaging facility so that no additional transportation is required. Alternately, the packaging facility could be located elsewhere. Subseabed would probably require two additional transport operations--transfer of waste packages to the embarkation port and subsequent ocean transport to the disposal site. Island repositories would require one additional movement, from the receiving port to the repository and would thus be equivalent to space disposal which would be characterized by a maximum of four major transport links for high-level waste. A smaller number of links could result from appropriate coloca-

TABLE 6.2.6. Performance of Proposed Waste Management Concepts on Ten Performance Standards

	Radiological Standards	1,000-Year Containment	10,000-Year Isolation	Developmental Time	Scientific Breakthroughs	Predictive Capability	Industry Size	Fuel Cycles	Reactor Design	Ability to Correct or Mitigate Failure	Maintenance & Surveillance
Mined Repository	X	X	X	X	X	X	X	X	X	X	X
Very Deep Hole	X	X	X	X	X	X	X	X	X	No	X
Rock Melt	X	No	No	X	X	X	X	X	X	No	No
Island Mined Repository	X	X	X	X	X	X	X	X	X	X	X
Subseabed	X	X	X	X	X	X	X	X	X	X	X
Ice Sheet	X	X	No	No	X	No	X	X	X	No	X
Well Injection	X	No	No	X	X	X	X	X	X	No	X
Transmutation	X	NA	NA	No	X	X	No	X	X	X	X
Space	X	X	X	X	X	X	X	X	X	No	X

X = The concept appears to have the potential to meet this standard based on available evidence.
No = The concept does not appear to have the potential to meet this standard based on available evidence.
NA = This standard is not applicable to this concept.

tion of facilities. The failure of a launch vehicle presents a potential single mode failure for space disposal and rapid rescue from incorrect earth orbit would likely be required to prevent public exposure.

Although, based on present evidence, any of the concepts could probably be conducted with radiation doses no greater than those currently permitted in fuel cycle facilities, substantial differences in cumulative radiation exposure might exist among the concepts. The above analysis suggests the following order of decreasing preference among concepts based on relative radiological effects during the operational period: mined repository; very deep hole; island mined repository; subseabed; space.

Post-Operational Period

Based on present evidence, any of the five concepts compared here has the potential to perform satisfactorily in the post-operational period (Section 6.2.3). However, probabilities of satisfactory performance differ and will be used as the basis of this comparison. Factors to be considered in evaluating the post-operational radiological integrity include failure of engineered containment to perform as expected, failure of natural barriers to perform as expected, compromise of repository integrity by catastrophic natural events exceeding design standards, and compromise of repository integrity by inadvertent human activity. From the standpoint of all four considerations, space disposal probably would provide the greatest certainty of satisfactory waste isolation in the post-emplacement period. In addition, the probability of satisfactory containment for several hundred years is seen as equally likely for the remaining concepts (see Section 6.2.3) although the performance of the package in the very deep hole is somewhat uncertain. Thus this discussion will focus on the prospects for longer-term isolation.

The effectiveness of natural barriers is seen to be potentially the greatest for the very deep hole concept because of the extreme depths involved. This assumes that depth alone will provide the single most effective barrier; however, uncertainties regarding the long-term integrity of the hole seal remain to be resolved. The mined repository concept relies on shaft seals as a barrier also but appears to offer greater probability of satisfactory long-term integrity due to the ability for human access during sealing operations. The possibility of disturbing the stability of the host sediment by emplaced waste might render the performance of the subseabed option less than that of mined geologic. The lack of understanding regarding behavior of island hydrologic systems under natural or waste-perturbed conditions raises significant questions as to the performance of the island mined repository in the long-term. For this reason the island mined repository concept is considered to be the least acceptable of the concepts on the basis of potential performance of natural barriers.

Of the four non-space concepts, very deep hole appears on the basis of its remote depth to offer superior protection from catastrophic natural events. Little distinction on this basis can be made between the subseabed, and mined repository concepts. Mined repositories on islands appear susceptible to catastrophic natural events associated with changes in future ocean levels.

As discussed in Section 6.2.1, efforts would be made to avoid siting repositories in areas having known or potential resource value, reducing the motivation for human intrusion. Fresh ground water can be a valuable resource in an island environment, however, and the presence of fresh water is intrinsic to the most potential island locations. Metal-bearing nodules are found--though they are scarce and of low grade--in the section of the ocean being considered for subseabed disposal. The resulting order of decreasing preference relative to prospects for inadvertent human intrusion would be space, very deep hole, mined repository, subseabed and island.

This overall analysis suggests the following order of decreasing preference relative to prospects for satisfactory radiological performance in the post-emplacement period: space; mined repository; very deep hole; subseabed; island mined repository.

6.2.4.2 Non-Radiological Environmental Effects

Health Effects

Implementation of any of the concepts would involve high-risk construction and operation activities including mining operations at sea and operations in space. Industrial accidents will undoubtedly occur; however, insufficient evidence currently exists to establish significant differences between options.

Injuries to the public could result from transportation accidents, and based on the number of transportation links inherent in each concept to which the public would be exposed (see Section 6.2.4.1), the order of decreasing preference would be the mined repository/very deep hole, island, and subseabed/space concepts. The mined repository and very deep hole concepts are essentially equivalent in this regard, as are the island and subseabed concepts.

Socioeconomic Effects

A comparative analysis of socioeconomic effects of generic disposal options is difficult because of the site specific nature of those effects. While one can assess factors such as size and number of facilities, the types of location and the size, timing and stability of the associated work force as discriminators among technology options, this is only half of the necessary information to assess impact. The other half consists of those factors associated with the area's ability to absorb the impacts. For example in times of high employment (no labor surplus) and high housing occupancy rates (no available housing) a project which requires high levels of manpower will create a serious (negative) impact. At a time when unemployment is high and housing is available, the same project would be of a positive impact.

Since these technologies involve different types of location and transportation steps, comparison against a "generic" location is not really possible. The addition of effects across several locations is not clearly a meaningful exercise since the impacts do not summate for any given community or person.

The mined repository and very deep hole disposal option would require only packaging plant and colocated repositories. Subseabed disposal would require a port facility in addition to packaging plants and the island concept would require, in addition, a receiving port and the island repository. The space disposal option would require processing, packaging, and launch facilities. An auxiliary waste disposal system for remotely handled and contact-handled TRU waste would likely be required for all concepts except mined geologic and island repositories.

In general, construction activities near small communities impact the socioeconomic structure of the community more than construction activities near large communities. Major facilities for the island geologic and subseabed disposal options would be located near the sea coast where the work force could typically be drawn from nearby communities. For the space disposal option, launch pad facilities exist and the required auxiliary facilities could be constructed at the launch site; however the waste treatment facility would also be required. The mined repository and very deep hole repositories would be located in areas of the continental United States, possibly in remote low population areas. In the case of space disposal especially there will likely be a substantial long-term increase in local employment due to the number of people required for support of launch activities. Subseabed has the same characteristics to a lesser degree, as does island disposal.

In conclusion, insufficient evidence (on a generic basis) is currently available to permit meaningful evaluation of alternative concepts on the basis of socioeconomic factors.

Aesthetic Effects

Aesthetic effects include noise, odors, and visual impacts. Analysis of aesthetic effects requires site-specific data because the effects are quite localized and dependent upon the design and siting of facilities. Because of this, characterization and comparison of aesthetic effects is not attempted in this Statement. Aesthetic effects would be an appropriate consideration in a statement considering proposed facility construction at a specific location. Items such as spoil piles from mined repositories and mud ponds from deep hole drilling could be unsightly, but the impacted area is not large.

Ecosystem Effects

Potential impacts of waste management facilities on ecosystems include effects on productivity, stability, and diversity. Evaluation of these effects at the generic level is difficult because of the sensitivity of these primary impacts to site and design characteristics which can only be addressed when considering specific installations. Consideration of such siting or design characteristics is beyond the scope of this generic statement. Thus to assess potential effects of the waste management options on ecosystems, it is necessary to look for effects inherent in the concepts under consideration.

Potential effects of the mined repository option include preemption of habitat during construction and operation of waste processing and repository facilities, potential releases of toxic waste processing chemicals to the environment and potential release of toxic spoil materials. Some preemption of habitat is unavoidable but with appropriate location and design might well be limited to a few hundred acres of low productivity habitat. Release of toxic materials presents a potentially more severe problem. While it is predicted that release of chemicals from waste packaging facilities can be controlled to acceptable levels, control of spoils may prove difficult because of the open air storage required.

Very deep hole repositories would produce ecosystem effects similar to the mined repository option. Spoils, however, would be less bulky and presumably easier to control.

Island geologic, though technically similar to the mined repository concept, has a greater potential for ecosystem disruption because of the sensitive and unique characteristics of many island ecosystems. Assuming careful design and management of such a facility, however, the facility exclusion area might well protect or restore the integrity of the natural ecosystem as has happened to some extent at the sites such as the DOE site near Hanford, Washington. Leach of the spoil pile could significantly effect the quality of a small island ecosystem.

The potential ecological effects of the subseabed option are not known at this time. On-shore facilities are likely to be constructed near populated (and presumably ecologically disturbed) areas because of current efforts to protect what remains of natural coastline. A large area of seabed would be subject to penetrometer emplacement; however, the population and productivity of the affected region is likely to be low and relatively minor disturbance would be experienced.

Ecological effects of space disposal are likely to be modest (with the exception of those normally associated with space flight launches) in comparison to the other options. Assuming space disposal of all high-level waste, ancillary geological repository requirements would be very small compared to disposing of all waste in terrestrial repositories.

All concepts under consideration here offer the potential for satisfactory performance on the basis of non-radiological environmental effects; however, important differences in the absolute magnitude of these effects may exist. Some discrimination is possible on the basis of non-radiological health effects to the general public; however, the generic nature of the study and the early stage of development of most of the concepts provide tenuous discrimination among concepts on the basis of occupational (non-radiological) health effects and socioeconomic, aesthetic, and ecological effects. The order of decreasing preference based on available evidence regarding non-radiological environmental effects is: mined repository/very deep hole, subseabed/island, space.

6.2.4.3 Status of Development

Availability of Technology for Construction of System

There are considerable differences among the concepts with respect to the engineering development needed for implementation. Construction for the mined repository and island repository options would use well-tested existing technology, although for novel applications. The waste treatment technology required to support the mined repository concept is also well advanced, having been the focus of substantial development. Less is understood relative to waste treatment and packaging requirements for an island mined repository, and considerable development activity might be required if the waste form and package concepts developed for mined repositories proved unsuitable for the island repository environment. The island concept would also require development of ocean transport and related transshipment facilities. Development of this equipment, however, is not viewed as particularly difficult, but largely an extension of existing technology.

The technology and methodology for siting geologic and subseabed repositories are developed to the point that they may be implemented. Space is unique in that the final location for disposition is not severely restricted by terrestrial concerns. Other options are poorly developed with respect to siting technology.

Implementation of the subseabed option, in addition to requiring development of the transshipment and ocean transport technology, would also require development of emplacement and emplacement monitoring technology, suitable waste form and packaging for the subseabed environment, and recovery technology for emplaced waste packages.

Space disposal would require development of a number of supportive technologies. Some (e.g., the space shuttle) are currently under development for other purposes and much of the remaining hardware represents extension of existing technology.

The very deep hole concept would require a significant extension of existing technology if the 10,000-m depth is required. Of the techniques available for making deep holes only rotary drilling has been used to develop wells to depths approaching those envisioned for very deep holes. Rotary drilling has been used for drilling to depths of about 9,000 m at bottom diameters of 6-1/2 inches--both shallower and of less diameter than postulated for the reference very deep hole concept. Deeper holes of larger diameter are thought possible but have not been demonstrated. It is quite possible that 10,000-meter holes will not be required by the concept. Other current limitations include casing to required depths and tensile strength of wire rope. In addition to technology related to making the very deep hole, development of a suitable waste form and packaging is required.

<u>Availability of Technology for Adequate Performance Assessment</u>

All of the alternative options appear to require further development of performance assessment and integrated safety and reliability analysis; however, the extent of such development is likely to be far greater with those concepts which have not received substantial attention, especially very deep hole, island mined repository, and space disposal. Fewer performance uncertainties appear to be associated with the subseabed concept; considerable research is underway on the deep ocean environment and the sediments are a homogeneous and probably fairly predictable environment. Fewest uncertainties appear to be associated with the mined repository concept largely because of the greater amount of research that has been accomplished on this concept.

The following order of decreasing preference is suggested relative to the current status of development of the concepts: mined repository; subseabed/island mined repository; space/very deep hole.

6.2.4.4 Conformance with Federal Law and International Agreements

The mined repository and very deep hole concepts could be developed without apparent conflict with Federal law or international agreements. A conflict may arise for the island disposal concept depending upon the island location. It would appear appropriate that the island be a possession of the U.S. Transport of large quantities of waste over international waters has the potential of generating adverse response.

Potential conflict of the subseabed disposal with existing law has been examined in some detail. The dumping of high-level radioactive waste is prohibited by the U.S. Marine Protection, Research and Sanctuaries Act of 1972, and therefore, would require Congressional action for implementation. The London convention of 1972, a multinational treaty on ocean disposal, addresses the dumping of contact-handled TRU and non-TRU waste. Dumping of high-level waste is prohibited; however the treaty's prohibition against dumping arguably does not extend to controlled emplacement of high-level waste into submarine geologic formations. EPA interprets the treaty as making subseabed disposal illegal.

Certain aspects of space disposal are addressed by existing treaties. The 1967 "Treaty on Principles Governing the Activities of States in the Exploration and Use of Outer Space Including the Moon and Other Celestial Bodies" prohibits waste disposal on the moon but does not rule out waste disposal in heliocentric orbit. Nations may object to the space disposal option because the waste would travel over their territory before being propelled from earth orbit. The 1972 "Convention on International Liability for Damage Caused by Space Objects" defines the responsibility for objects falling to earth on other countries. Consideration of such liability would be required.

In summary, the decreasing order of preference emerging from consideration of possible legal constraints on implementation of the five concepts is: mined repository/very deep hole; island; space; subseabed.

6.2.4.5 Independence from Future Development of the Nuclear Industry

Of the five concepts under comparison, space disposal appears to be most sensitive to the future development of the nuclear industry since it is considered that a substantial nuclear capacity will be required to justify the required investment (Section 6.2.3).

6.2.4.6 Cost of Development and Operation

Preliminary estimates of the cost of construction and operation for the mined repository, very deep hole and subseabed concepts appear in Section 6.1. These have been compiled and converted to unit costs (mills/kWh) in Table 6.2.7. Cost estimates for the island mined repository and the space disposal concept were insufficiently complete to permit reduction to a unit basis.

Of the available unit cost estimates, the very deep hole concept appears to be the most expensive with estimated costs of 3.0 mills per kilowatt-hour (1980 dollars), not a significant proportion of typical current new construction power costs (30 to 50 mills/kWh). Because these cost estimates are very preliminary and because even the most costly option

TABLE 6.2.7. Estimated costs of Various Disposal Options (1980 dollars)

	Research and Development Cost $ millions	Pre-Disposal Cost, $/kgHM		Repository Costs			Total Cost, mills/kWh[a,b,c]	
		Once-Through	Reprocessing	Construction, $ millions	Operating, $ millions/year	Decommissioning, $ millions	Once-Through	Reprocessing
Mined Repository, 6,000 MTHM/yr	3,700	100	170	2,600	87	25	0.7	1.0
Very Deep Hole, 5,000 MTHM/yr	900	100	170	2,800	2,100	40	2.5	3.0[d]
Island	NA[e]	150	190	NA	NA	NA	NA	NA
Subseabed, 5,000 MTHM/yr	NA	150	190	760	29	54	0.8	0.9
Space, per flight	NA	210	170	NA	46[f]	4	NA	NA

(a) Does not include Research and Development costs.
(b) Construction and decommissioning costs amortized over 17 years @ 7%.
(c) Waste production rate is 38 MTHM/GW-year.
(d) Includes 0.2 mills per kWh for ancillary repository.
(e) NA = not available.
(f) $ million per flight.

appears not to significantly impact the cost of electrical power, a cost comparison should not currently be assigned significant weight in this analysis. It should be noted that the cost estimates for all concepts essentially assume that no currently unanticipated questions will arise, which is probably an unlikely assumption.

6.2.4.7 Potential for Corrective or Mitigating Action

Prior to closure and sealing of access tunnels and shafts, mined repositories (including those utilized in the island disposal concept) would allow failure detection and permit retrieval of waste canisters. This system allows flexibility to future generations as to how long they might choose to leave the facilities open to inspection. Following closure, failure detection would be more difficult, although remote instrumentation could be installed for this purpose. Corrective action would be difficult (though possible) as the location of the waste would be known and access tunnels could be reopened. Detection of repository failure exemplified by unexpected concentrations of radionuclides could allow the mitigating actions of restriction of access to contaminated aquifers and other measures including evacuation of affected areas.

Complete corrective action capability for the island mined repository concept would require development of systems for locating and retrieving casks lost at sea in the case of the sinking of a transfer ship. A similar system would be required for the subseabed concept. Transponder devices would be fitted to the casks while enroute, and location and retrieval of an individual cask from the seafloor is considered feasible using existing equipment. However, loss of a ship with waste within the hull would severely complicate retrieval operations. Retrieval of emplaced canisters is considered to be feasible using existing overcoring technology, although retrieval of a large number of canisters would likely be very expensive.

Full corrective action capability for space disposal would require a deep-ocean payload retrieval system if system failure released radionuclides to the atmosphere. No corrective action would be possible. If failure of the space disposal system were to occur after achieving orbit, backup launch and orbit transfer vehicles, and some means for correction of improper orbit would be required. Each of these is under consideration as part of the space disposal concept, and if successfully developed (along with appropriate monitoring systems), would provide corrective action capability for most situations.

Corrective action with the very deep hole concept is thought possible only while the package is attached to the emplacement cable.

In summary, mined repositories appear to offer the greatest potential for corrective action. Subseabed appears also to provide reasonable potential for corrective action with the principal problem being retrieval of waste from a transport ship lost at sea. Island mined repositories present the combined difficulties and assets of the subseabed and mined repository concepts. Full corrective action potential appears to be achievable with space disposal for all situations except failure of the waste packaging system during launch or pre-orbital operations. Corrective action is thought not to be possible with the very deep hole concept following package disengagement. The following order of decreasing preference relative to corrective action is thus suggested: mined repository; island mined repository; subseabed; space/very deep hole.

6.2.4.8 Long-Term Maintenance and Surveillance Requirements

None of the five concepts being considered here appear to require significant maintenance and surveillance activities during the post-operational period.

6.2.4.9 Resource Consumption

Preliminary estimates of selected critical resources for mined repository, very deep hole, subseabed and space disposal are provided in Table 6.2.8. Because of the very preliminary state of development of most concepts as reflected in the apparent inconsistencies among the estimates of Table 6.2.8, comparisons on the basis of these estimates would not be meaningful.

6.2.4.10 Equity of Risk

None of the concepts appear to have significant differences in this respect. Subseabed, ice sheet, island, and space disposal have the positive feature that no one must live in close proximity to the final disposal location. This creates the initial impression that the impact and risk are far less for those alternatives than for mined repositories. However a situation is established wherein the process of transportation of wastes is channeled through one location. A judgement of the equity of risk and impact resulting from the focus of transportation versus the focus of disposal is yet to be established.

6.2.5 Conclusions

Results of the comparisons on the assessment factors are depicted in Table 6.2.9 which shows the preference rankings of the five concepts (mined repository, very deep hole, subseabed, island repository, and space) on each of the assessment factors for which discrimination was found among the concepts. For each factor, the rankings of the five waste management concepts are plotted along a preference continuum, ranging from "most preferred" at the extreme left to "least preferred" at the extreme right. Concepts are clustered where no differences were observed.

6.2.5.1 Mined Repository

Examination of Table 6.2.9 supports selection of the mined repository concept as the waste disposal concept for preferred development. This concept is a "most preferred" concept on six of the seven comparisons of Table 6.2.9, ranking second on one consideration, "Radiological Effects During the Post-Operational Period." Here, the apparent length of isolation provided by space disposal results in the latter being preferred to mined repositories. An overall evaluation of the Radiological Effects attribute, however, might place

TABLE 6.2.8. Estimated Resource Commitments for Various Repositories

Critical Resource	Mined Repository[a,c]	Very Deep Hole[b]	Subseabed[b]	Space[b]
Aluminum, MT	220	13,000	13,000	83,000
Chromium, MT	--	14,000	14,000	5,000
Nickel, MT	--	7,500	7,500	2,000
Water, m	1,300,000	199,000,000	--	60,000,000
Natural Gas or Propane, m	11,500	10,000,000	10,000,000	10,000,000
Electricity, kWh	3,400,000,000	56,000,000,000	20,000,000,000	59,000,000,000
Petroleum-Derived Fuel, m^3	5,300,000	6,000,000	5,100,000	1,500,000
Other Fuel, MT	--	--	--	4,800,000

(a) Highest consumption construction scenarios of Tables 5.4.2 and 5.4.3 added to operational values.
(b) Highest consumption scenario indicated of Section 6.1.
(c) Island mined repository has similar commitments.

TABLE 6.2.9. Summary of Preference Rankings

	Most Preferred ⟶ Least Preferred
Radiological Effects	
Operational Period	(MR) . . . . (VDH) . . . . (IMR) . . . . (SS) . . . . (S)
Post-Operational Period	(S) . . . . (MR) . . . . (VDH) . . . . (SS) . . . . (IMR)
Non-Radiological Environmental Effects	(MR, VDH) . . . . (SS, IMR) . . . . (S)
Status of Development	(MR) . . . . (SS, IMR) . . . .(S, VDH)
Conformance with Law	(MR, VDH) . . . . (IMR) . . . . (S) . . . . (SS)
Independence from Future Development of the Nuclear Industry	(MR, VDH, IMR, SS) . . . . (S)
Potential for Corrective or Mitigating Action	(MR) . . . . (IMR) . . . . (SS) . . . . (S, VDH)

KEY: MR = Mined Repository
VDH = Very Deep Hole
IMR = Island Mined Repository
SS = Subseabed
S = Space.

space disposal in an intermediate position below mined repositories because of the low ranking of space disposal on the basis of radiological effects during the operational period.

6.2.5.2 Subseabed

No clear preference emerges between the subseabed disposal concept and the island mined repository concept. However, because of significant uncertainties regarding the long-term radiological integrity provided by island geologic and hydrologic systems, subseabed appears to be superior to the island mined repository concept for continued development as an alternative to mined repository waste disposal. An additional advantage may be provided by subseabed's unique characteristics as a genuine conceptual alternative to mined repositories in comparison with island disposal, which is basically a variant (with additional uncertainties) of the mined repository concept. Uncertainties remain to be resolved concerning the long-term integrity of the emplacement media; development of transportation, emplacement and monitoring technology; resolution of potential international conflicts; and development of corrective action capabilities. Research will still be required, especially with the objective of resolving the waste isolation potential of the subseabed sediment. Should this capability be demonstrated conclusively, engineering development of the system could proceed.

6.2.5.3 Very Deep Hole

Although not possessing any clearly defined advantages over the mined repository concept on the basis of currently available evidence, the very deep hole concept ranks generally high on most of the assessment properties. Very deep hole offers potential for a high degree of geologic barrier performance in the post-operational period and some possibility of superior working conditions compared to mined repositories. A key issue is the value of manned in-situ examination of the actual placement location to understand the condition and environment into which the waste package is to be placed. Significant problems remain however, including the need for substantial development of drilling technology, improved understanding of the geologic environment at very deep hole depths, and analytical verification of the postoperational integrity of very deep hole repositories and performance of packages at the requisite temperature and pressure. Since deep hole technology is being developed for other reasons (e.g., for geopressured methane and for geothermal purposes) it is likely that increased information will be available regarding these uncertainties. An additional problem is the difficulty of providing adequate corrective action capability. Thus, the very deep hole concept, though having potentially superior characteristics to other alternatives, is also characterized by greater uncertainties. For these reasons, although continued development of the very deep hole concept as a long-term alternative to mined repositories is recommended, the priority of development is considered to be secondary to the subseabed concept. The considerations of potential problems with corrective action and the relatively unadvanced status of technology weigh heavily in this decision.

6.2.5.4 Space Disposal

The principal argument for space disposal is its promise for extraterrestrial disposal of selected radioisotopes; but substantial reservations exist concerning this concept. These include the potential radiological risk of the concept during the operational period, non-radiological health effects, potential conflicts with international law, and the difficulty of developing acceptable corrective action capabilities. Because of these conditions, priority development of space disposal as an alternative to mined repositories would appear to be unwise.

6.2.5.5 Island Disposal

The island disposal concept appears to present few advantages over the subseabed concept or the mined repository and is characterized by significant uncertainties regarding its potential for long-term isolation of waste. The principal potential advantage of island disposal is sociopolitical--it offers the possibility of a repository site remote from habitation and, thus, possibly of greater acceptability to the general public. Furthermore, the potential for international cooperation in establishing a repository at a "neutral" site might be presented by an island. Subseabed, however, offers the same advantages; thus the island concept would have merit only if the sociopolitical advantages were seen to be highly important, an appropriate island were available, and if the subseabed concept proved not to be technically acceptable. Because of these considerations, and because of great uncertainties regarding the waste isolation potential of island geology, development of this concept is not recommended.

REFERENCES FOR SECTION 6.2

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CHAPTER 7

SYSTEM IMPACTS OF PROGRAM ALTERNATIVES

To assess and compare the impacts of implementing the three program alternatives addressed in this Statement (see Section 3.1), an analysis was made using a computer simulation of the complete waste management system functioning over the lifetime of a nuclear power system. This analysis considers the treatment and disposal of all post-fission high-level[(a)] and TRU wastes (including decommissioning wastes), as well as gaseous and airborne wastes. All waste management functions are accounted for and all radioactive waste streams are tracked each year from origin through treatment, storage, transport and accumulation in a disposal repository. Both the example once-through cycle and the example reprocessing cycle described in Section 3.2 and Chapter 4 are analyzed.

7.1 BASIS FOR SYSTEM SIMULATION

To cover the range of potential impacts of program implementation, five different nuclear power growth cases are considered. In all cases, the nuclear capacity is assumed to consist of one-third BWRs and two-thirds PWRs. These cases were described in Section 3.2 and can be summarized as follows.

Case 1--Present Inventory. In this case, we consider only the amount of spent fuel estimated to be on hand, including in-core fuel, at the end of 1980; this is approximately 10,000 MTHM.

Case 2--Present Capacity. In this case, we consider the amount of spent fuel that would result from continued operation of the present 50 GWe of nuclear capacity over its expected normal life cycle to retirement after 40 years operation.

Case 3--250 GWe in Year 2000. In this case, nuclear power capacity grows to 250 GWe in the year 2000. All nuclear power plants operate for an expected normal life cycle of 40 years, and the last plant shuts down in 2040. It is intended to assess the waste management impacts over the complete life cycle of a nuclear generating system.

Case 4--250 GWe Steady State. This case follows the same growth curve, to 250 GWe in the year 2000, but then replaces retired capacity to maintain the 250 GWe capacity to the year 2040 when the case terminates.

Case 5--500 GWe in Year 2040. In this case, we assume the same 250 GWe growth by the year 2000 as in Case 3 but continue capacity additions to 500 GWe in the year 2040 when the case terminates.

The nuclear capacities for these cases are shown in Table 3.2.1 and Figure 3.2.3. The total electric energy generated in these five cases is shown in Table 7.1.1. Although power generation terminates in the year 2040 in all cases, waste management operations and decommissioning activities are continued until all wastes are emplaced in disposal facilities. In all cases, this is accomplished by the year 2075. The system simulation encompasses a

(a) High-level waste in this context includes spent fuel in the once-through cycle.

TABLE 7.1.1 Electric Energy Generated in Nuclear Power Growth Scenarios

Case	GWe-Yr
1	200
2	1,300
3	6,400
4	8,700
5	12,100

period from 1980 to 2075. In addition, the radioactivity inventory in the final repositories is followed over a million-year period. This provides an accurate representation of the radioactivity source term for hazard analysis. However, because of the very large uncertainties associated with long-term predictions of events that might result in some future radiological hazard, it is not considered useful to attempt predictions of radiological consequences for periods beyond about 10,000 years.

The objective of the system simulation was to identify the cumulative impacts of implementing the proposed program and to compare the range of impacts that would result from implementation of the proposed program, with those that could result from implementation of the alternative program or the no-action alternative. The three program alternatives were described in Section 3.1 and can be summarized as follows.

- Proposed Program. The research and development program for waste management will emphasize use of mined repositories in geologic formations capable of accepting radioactive wastes from either the once-through or reprocessing cycles. This program will be carried forward to identify specific locations for the construction of mined repositories.
- Alternative Program. The research and development program would emphasize the parallel development of several disposal technologies. This action implies an R&D program to bring the knowledge regarding two or three disposal concepts and their development status to an approximately equal level. At some later point, a preferred technology would be selected for construction of facilities for radiological waste disposal.
- No-Action Alternative. This alternative would eliminate or significantly reduce the Department of Energy's research and development programs for radioactive waste disposal. Under this alternative, existing spent fuel would be left indefinitely where it is currently stored and any additional spent fuel discharged from future operation of commercial nuclear power plants would likewise be stored indefinitely in water basin facilities either at the reactors or at independent sites.

The proposed program represents adoption of the interim planning strategy referred to in the President's statement of February 12, 1980, announcing a comprehensive radioactive waste management program for this nation. The President stated in part, "I am adopting an interim planning strategy focused on the use of mined geologic repositories capable of accepting both waste from reprocessing and unreprocessed commercial spent fuel." Final adoption of this strategy was to be subject to "a full environmental review under the National Environmental Policy Act" which this Statement satisfies. The President further stated, "We should be ready to select the site for the first full-scale repository by about 1985 and have it operational by the mid-1990s." Subsequent to the President's statement the Department of Energy published (on April 15, 1980) a Statement of Position on a proposed NRC rulemaking on storage and disposal of nuclear waste (DOE/NE-0007). DOE states in that document that implementation of the interim waste disposal strategy will result in the establishment of operating geologic repositories within the time range of 1997 to 2006. An exact date of operation, depending on a number of variables, will be determined by the outcome of existing programs. For example, if a site in bedded or domed salt is selected and licensing schedules recently forecast by the NRC staff are assumed, repository operation as early at 1997 could be achieved. However if a hard rock such as granite is selected, and if allowances are made for other uncertainties such as licensing proceeding delays and a requirement for more rigorous subsurface site characterization prior to site selection, initial repository operation could be as late as 2006. To cover additional contingencies such as an accelerated effort to open a repository or, at the other extreme, additional delays for reasons not yet foreseen, a range of repository startup dates from 1990 to 2010 is used here. The range of impacts is important in this simulation rather than the specific dates of repository startup.

Implementation of the alternative program would result in extending the time to operation of the first disposal system. This action implies a further period of research and development to bring the development status of the selected disposal alternatives to an approximately equal status with current knowledge regarding geologic disposal. At that time, a preferred technology would be selected and effort would be concentrated on developing this preferred technology with a program similar to the currently planned program for implementing geologic disposal. Thus a substantial time delay is inherent in this alternative.

In this system simulation, mined geologic repositories are used to represent the disposal method ultimately selected under the alternative program. This concept is the only one developed sufficiently to model impacts and costs reasonably well, and any alternative disposal concept that might be selected would only be selected if it did not have significantly greater impacts or costs. The primary effect of the alternative program implementation is the required interim storage for spent fuel or reprocessing wastes, the additional transportation to and from this storage and the impacts and costs for these operations. Benefits of the delay inherent in this alternative program include the processing and disposal of older and thus less radioactive and cooler wastes. Implementation of this alternative program is simulated by a range of repository startup dates from 2010 to 2030.

For the no-action alternative, indefinite storage of spent fuel in water basin facilities with no ultimate disposal has been assumed. It is also assumed that reprocessing would not be undertaken. Only the first three nuclear growth cases are considered because, without disposal, growth of nuclear power generation beyond the year 2000 does not appear credible.

The nuclear power growth cases and repository startup dates considered for the once-through cycle system simulation are shown in Table 7.1.2. A range of repository startup dates was used for the first three cases, that is, 1990 to 2010 representing the proposed program and 2010 to 2030 representing the alternative program. The 2010 startup provides both the last year of the range under the proposed program and the first year of the range under the alternative program. To simplify the analysis, only a single mid-range repository startup date, year 2000 representing the proposed program and 2020 representing the alternative program, was used for Cases 4 and 5. However, the same potential range as in the other cases should be inferred.

The nuclear power growth cases and reprocessing and repository startup dates considered for the reprocessing system simulation are shown in Table 7.1.3. Cases 1 and 2 were eliminated from consideration here because reprocessing was not considered to be credible under

TABLE 7.1.2. Repository Startup Dates Considered in the Once-Through-Cycle System Simulations

Nuclear Power Growth Cases	Proposed Program	Alternative Program	No-Action Alternative
1. Present Inventory Only	1990 to 2010[a]	2010[a] to 2030	None
2. Present Capacity Normal Life	1990 to 2010[a]	2010[a] to 2030	None
3. 250 GWe System by Year 2000 and Normal Life	1990 to 2010[a]	2010[a] to 2030	None
4. 250 GWe System by Year 2000 and Steady State	2000	2020	--
5. 500 GWe System by Year 2040	2000	2020	--

(a) These cases are identical under both the proposed and alternative programs.

TABLE 7.1.3. Reprocessing and Repository Startup Date Combinations Considered in the Reprocessing-Cycle System Simulations

Nuclear Power Growth Cases	Proposed Program Reprocessing	Proposed Program Repository	Alternative Program Reprocessing	Alternative Program Repository
3. 250 GWe System by Year 2000 and Normal Life	1990	1990	1990	2010[a]
	1990	2010[a]	2010	2010[a]
	2010	2010[a]	1990	2030
			2010	2030
4. 250 GWe System by Year 2000 and Steady State	2000	2000	2000	2020
5. 500 GWe System by Year 2040	2000	2000	2000	2020

(a) These cases are identical under both the proposed and alternative programs.

these low-growth conditions. The reprocessing cases are complicated by the added uncertainty for reprocessing startup. For Case 3, reprocessing startup in the time period 1990 to 2010 was considered in combination with repository startup dates of 1990 to 2010 for the proposed program and repository startup dates of 2010 to 2030 for the alternative program. As in the once-through cycle cases, the 2010 repository startup provides both the last year of the range under the proposed program and the first year of the range under the alternative program. To simplify the analysis, only mid-range dates were considered for Cases 4 and 5, that is, reprocessing startup in year 2000 in combination with repository startup in year 2000 representing the proposed program and in year 2020 representing the alternative program. However, the same potential range as in Case 3 should be inferred.

In selecting reprocessing startup dates, it was assumed that even if the current moratorium on reprocessing were lifted immediately, at least 10 years would be required to complete the construction, licensing, and startup of a reprocessing facility. Since a considerably longer time period could conceivably be required before reprocessing could be initiated, the 2010 startup date was selected to illustrate the effect of reprocessing after a longer period of delay. The important factor here is not the reprocessing dates themselves, but the effect that a range of reprocessing startup dates has on waste management impacts.

7.2 METHOD OF ANALYSIS FOR SYSTEM IMPACTS

The information flow in the computer simulation used for this analysis is presented in Figure 7.2.1. The first two modules of this computer model (i.e., ORIGEN and ENFORM) were adaptations of existing programs (Bell 1973, Heeb et al. 1979), while the last two modules were developed specifically for this simulation.

The computer code ORIGEN (Bell 1973) was used to define spent fuel composition. The ORIGEN code calculates the average composition of the spent fuel discharged from a nuclear reactor based on a set of input parameters that characterize the irradiation conditions. The set of input parameters (i.e., neutron cross sections and spectral indices) used had been calibrated to match results of empirically measured spent fuel compositions. Isotopic data were calculated for 175 nuclides, including all significant fission products, activation products and actinides.

Twenty-eight ORIGEN cases representing both PWR and BWR fuel irradiations were used to describe the spent fuel compositions for all of the fuel cycle alternatives. These cases (see DOE/ET-0028, Sec. 10.1) include separate cases for each enrichment zone of the initial cores, a first reload and equilibrium reload fuel batch and three recycle fuel batches for both uranium and plutonium recycle. In addition, the low exposure fuel batches remaining when a plant is shut down for decommissioning are described. Whether recycling is used or not, all plants start up and shut down without recycle fuel in the core. Recycle of both uranium and plutonium is limited to equilibrium fuel reloads, and the amount of either recycle fuel in any year is limited to 50% of the equilibrium reload fuel.

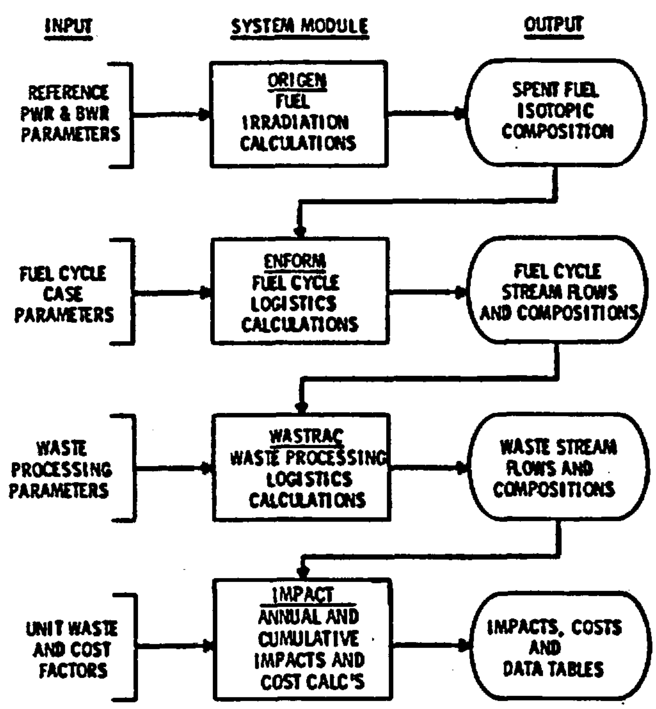

FIGURE 7.2.1. System Simulation Information Flow

By combining the ORIGEN to match the annual operating status of all plants in the system and the amount of uranium and plutonium available for recycle, the spent fuel composition with or without recycle in any year can be determined. This method of using a relatively small number of fuel irradiation (burnup) calculations to characterize a large number of spent fuel combinations provides an efficient and reasonably accurate representation of spent fuel compositions each year for the entire system.

The number of recycles for both uranium and plutonium was limited to three. The amount of third-recycle uranium and plutonium is small and the accumulation of ^{242}Pu in the third-recycle plutonium discharge reduces its value substantially. For these reasons and to simplify the calculation, the discharge from third-recycle fuel was discarded. In a real system whether or not the plutonium from the third recycle would be recycled would most likely be an economic decision. It could continue to be recycled and ultimately either be fissioned or transmuted to higher actinides and be discarded in the waste.

The computer code ENFORM (Heeb et al. 1979) was used to develop fuel cycle logistics and isotopic compositions of the fuel cycle streams. ENFORM was originally developed to evaluate environmental impacts of the entire nuclear fuel cycle. However, only its fuel cycle logistics capabilities were used here to provide fuel cycle source data for the WASTRAC module, which determined waste management logistics.

ENFORM input requirements include:

- a nuclear power growth projection
- a life-cycle operating schedule for the nuclear power plants
- recycle assumptions, i.e., once-through or recycle
- a fuel reprocessing schedule if recycle is selected
- inventory and timing assumptions for the entire fuel cycle
- spent fuel compositions as calculated by ORIGEN.

The output of the logistics calculation is a year-by-year mass flow and isotopic composition for each operation in the fuel cycle.

The computer code WASTRAC, developed for this analysis, models the storage, treatment, packaging, shipment and disposal operations for each waste stream. Figure 7.2.2 illustrates the waste management steps and the items calculated in a typical WASTRAC subsystem. Waste management steps can be added or deleted as required to model a specific subsystem. Each waste stream was tracked through a series of steps similar to that displayed in Figure 7.2.2.

WASTRAC computes waste volume and waste composition as a function of year, waste type and waste management step. The entire radionuclide content of the spent fuel is accounted for by allocating it either to a product stream, i.e., uranium or plutonium in a reprocessing case, or to one of the waste streams. Radionuclide inventories are corrected at each step for decay or buildup during the time interval since reactor discharge and/or reprocessing. Radionuclide inventories are also calculated for times up to one million years after placement in a final repository.

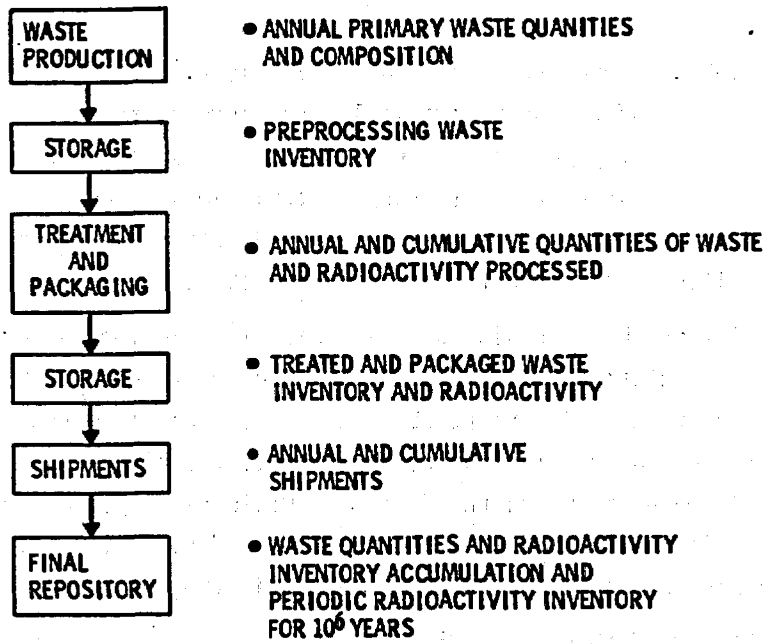

FIGURE 7.2.2. WASTRAC Calculations

The output of WASTRAC provides the waste volume and the quantity of each isotope in each waste stream at each step in the waste management system. Each treated waste stream is classified by container type and by the surface dose class for the treated TRU waste containers. Specifically waste streams are classified as high-level waste, remotely handled TRU (RH-TRU) waste (container surface-dose-rate equal or greater than 200 mrem/hr) or contact-handled TRU (CH-TRU) waste (container surface-dose-rate less than 200 mrem/hr).

The final step in the system simulation uses the time-dependent waste logistics data from WASTRAC to calculate the waste management impact and costs and to compile results in a series of tables. The computer code IMPACT was developed to perform these functions.

By utilizing release fractions for each isotope and each waste stream at each waste management step and dose factors per curie released, the isotopic releases and 70-year population radiation doses for each waste stream at each waste management step are calculated. Regional dose to whole body, bone, lungs, and thyroid and worldwide dose for release of ^{3}H, ^{14}C, and ^{85}Kr are calculated.

The IMPACT program organizes the results of the WASTRAC calculations, sums up annual and cumulative totals at specified intervals and prepares a series of tables to display the results. IMPACT also calculates both undiscounted and present-worth[a] costs as well as levelized[b] waste management costs per unit of power produced and per unit of fuel used.

(a) Present-worth discounting is a method of allowing for the time value of money. The present worth may be thought of as a present sum of money equivalent to a specified future payment or receipt or to a series of future payments or receipts. The present worth of a payment is obtained by multiplying the payment by $1/(1 + i)^n$, where i equals the interest rate or discount rate and n is the number of years from the present to the time of the payment. The present worth of a series of payment is obtained by summing each payment's present worth.
(b) Levelizing refers to developing a single, constant unit charge, which recovers an expenditure associated with a facility or system including interest (see Section 3.2.8.2).

Four types of waste management costs are computed including treatment, interim waste storage, transportation, and repository costs. All costs are based on estimated unit costs as described in Sections 4.9 and 5.6. The cost of high-level waste treatment reflects an adjustment of high-level waste volume per container as limited by the thermal criteria at the geologic repository and the thermal energy of the waste at the time of emplacement.

Figure 7.2.3 schematically illustrates the relationship between the cash flow of the individual waste management system components and the discounting procedures. There are two similar but distinctly different applications of discounting techniques used in the development of the equivalent electric power and fuel cost of waste management. First, a present-worth levelizing procedure is used to develop unit costs, i.e., cost per unit of spent fuel, for each waste management function. Second, a separate present-worth levelizing procedure is used to convert waste management costs to equivalent electric power and fuel costs.

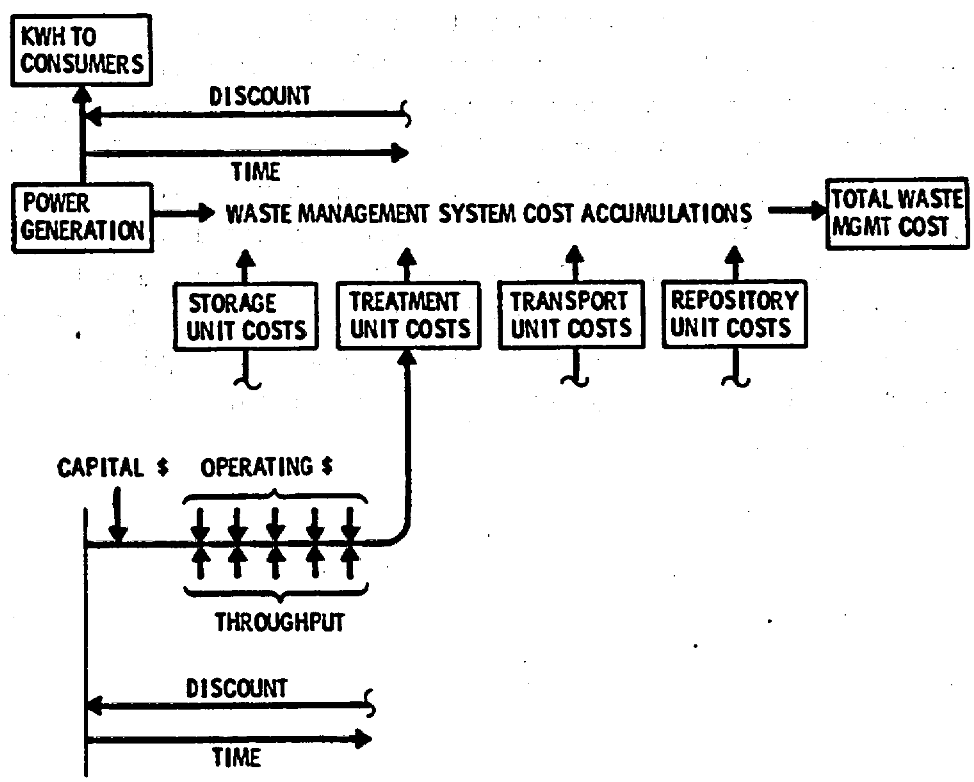

FIGURE 7.2.3. Time and Discounting Relationships of Waste Management Functions of Cost

The lower row of boxes in Figure 7.2.3 illustrates the functions that contribute to the total waste management system costs. The additional detail under the treatment unit-costs box indicates the flow of dollars and materials that are factored into the development of unit waste management costs. For any single waste management function all of the cash flows are present-worth discounted to a common starting point. The levelized unit cost for that function is then calculated by the relationship:

$$\text{Unit Cost} = \frac{(\text{Sum of present-worth costs})}{(\text{Sum of present-worth throughput})}$$

The unit cost developed by this procedure represents the single charge that can be assessed for the waste management function over the life of the facility that will recover all expenditures plus a return (the discount rate) on any unrecovered investment during the life of the facility. The sum of all the separate waste management system unit costs represents the total waste management system unit cost.

The accumulation of the waste management costs over a period of time following generation of power is also illustrated in Figure 7.2.3. It is assumed that all waste management costs, whether the services are provided by private industry or by the government, will be borne by the consumers of the electric energy generated by the nuclear power facility. Thus, the waste management costs will be reflected as an increase in cost of power.

The equivalent power costs of waste management can be obtained by discounting the costs of the individual waste management functions to the time of power generation, summing them all and dividing by the kilowatt hours of electric energy produced during the irradiation of the fuel. In other words, money is assumed to be collected from the rate payers to cover the cost of waste management at the time the electricity is generated. The amount collected is somewhat less, depending on the discount rate, than the costs of waste management will be when it is actually incurred. This allows the utility to earn a return on this money during this period so that a sufficient fund accumulates to pay for the waste management costs at the time they are incurred. At any interest rate (discount rate) greater than 0%, fewer dollars need be collected from the rate payers than will be required to pay later waste management costs at the time they are incurred. The higher the utility discount rate, the lower the waste management costs become.

7.3 SYSTEM LOGISTICS

To develop the system logistics requirements, some assumptions were made regarding the characteristics of a future nuclear industry and its associated waste management systems. These assumptions are not intended to be predictions of the future; rather, they are intended to provide a basis for estimating a potential range of requirements over a broad range of possible future developments. The results are valid primarily in terms of potential ranges of values. In general, the assumptions are intended to be conservative; that is, they err in a direction that tends to overstate rather than understate potential requirements and impacts.

The assumptions made in developing the logistics requirements for the once-through cycle were as follows.

1. Spent fuel is stored for a minimum of five years at the reactor basins after which it can be shipped to a repository if one is available.
2. The maximum storage capacity at the reactor basins averages 7 annual discharges. This is based on the assumption that reactor basin capacity will be expanded, on the average, to provide capacity for at least 3 full cores. Retaining full-core discharge capability and considering 3 annual discharges per core for a PWR and 4 annual discharges per core for a BWR results in an average capacity for approximately 7 annual discharges. This assumption also results in away-from-reactor storage requirements that approximate the maximum requirements shown in a recent study when currently licensed expansion plans of the electric utilities are assumed to be implemented and full-core reserve is maintained (DOE/NE-0002 1980).
3. After reactor storage basin capacity is filled, excess spent fuel is shipped to an away-from-reactor (AFR) independent spent-fuel storage facility.
4. When a repository opens, spent fuel is sent to the repository on a first-in, first-out basis; that is, the oldest fuel is always sent to the repository first.
5. Repository receiving capacity is expanded according to the following schedule for the first 10 years:

Year	1	2	3	4	5	6	7	8	9	10
Receiving Capacity, MTHM	700	1,300	2,000	2,000	2,000	2,700	3,300	4,000	4,000	4,000

   After 10 years 2,000 MTHM capacity increments can be added annually as needed to meet the demand. This capacity does not necessarily represent a single repository, but may represent several repositories that are opened up sequentially. However, single repositories with receiving capability of at least 6,000 MTHM per year are considered feasible.
6. The distance from a reactor to an AFR storage facility is 1,000 miles.
7. The distance from either a reactor or an AFR facility to a repository is 1,500 miles.

8. Spent-fuel from reactors is shipped 10% by truck and 90% by rail (45% by a combination of truck and rail using intermodal casks that can be transported by truck for short distances to a rail siding where they are transferred to a rail car and 45% by rail-only) while shipments from AFR facilities are 100% by rail.

The assumptions made in developing the logistics requirements for the reprocessing cycle were as follows.

1. A minimum storage period for spent fuel at the reactor basin is one year and at the reprocessing plant is one-half year.
2. The maximum storage capacity at the reactor averages 7 annual discharges.
3. Fuel that cannot be stored at the reactor basins is shipped to AFR storage facilities.
4. The reprocessing plant receives and processes spent fuel on a first-in, first-out basis; that is, the oldest fuel is processed first.
5. Reprocessing capacity is expanded in a pattern similar to the repository receiving capacity except that here each capacity increment is intended to represent a separate plant. Each plant has a 2,000 MTHM per year capacity and the second and third plants are restricted to startups at 5-year intervals. Each plant has a two-year restricted-throughput startup period, i.e., 700 MTHM in the first year, 1,300 MTHM in the second year and 2,000 MTHM/year thereafter. After 10 years, the interval between plant startups is restricted to a 3 year minimum.
6. Solidified high-level waste is stored for 5 years at the reprocessing plant before shipment. TRU wastes can be shipped as they are packaged.
7. If a repository is not available to receive the reprocessing plant wastes, storage is provided for high-level waste and TRU wastes at a separate independent site.
8. When the repository opens, it receives the wastes on the basis of the oldest waste first at the same rate they are produced. After 10 years, the receiving rate is accelerated as necessary to eliminate the storage backlog at the end of the 30th year.
9. When interim storage is required, all wastes flow through the storage facility until the backlog is eliminated. This assures that the oldest waste is sent to the repository first.
10. Shipping distances for spent fuel to the reprocessing plant or interim storage and from interim storage to reprocessing are 1,000 miles. Treated waste shipment distances from reprocessing or MOX fuel fabrication plants to interim storage are also 1,000 miles.
11. Shipping distances from the reprocessing or MOX fuel fabrication plants or from interim storage to a repository are 1,500 miles.

7.3.1 Repository Inventory Accumulations

The total amount of spent fuel to be disposed of or reprocessed for each of the five growth assumptions is shown in Table 7.3.1. The relative quantities of spent fuel here are approximately the amount that would result from the quantities of generated energy shown in Table 7.1.1. The proportional relationship is not exact, however, because only in Cases 2 and 3 do all reactor plants complete their full normal-life cycles.

TABLE 7.3.1. Total Spent Fuel Disposal or Reprocessing Requirements

Case	Nuclear Power Growth Assumption	Spent Fuel Discharged, MTHM
1	Present Inventory Only	10,000
2	Present Capacity and Normal Life	48,000
3	250 GWe System by Year 2000 and Normal Life	239,000
4	250 GWe System by Year 2000 and Steady State	316,000
5	500 GWe System by Year 2040	427,000

Only the once-through cycle is considered for the first two (low-growth) cases. The accumulation of spent fuel in the final repositories for these two cases is plotted in Figure 7.3.1 for each of the three repository startup dates. The region between the first two curves represents the range of inventory accumulations possible for the proposed program while the region between the second and third curve represents the range of inventory accumulations for the alternative program.

The repository inventory accumulation for Case 3 using the once-through cycle is shown in Figure 7.3.2. With the reprocessing cycle, however, the repository inventory accumulation is a function of both the reprocessing throughput and the repository startup and

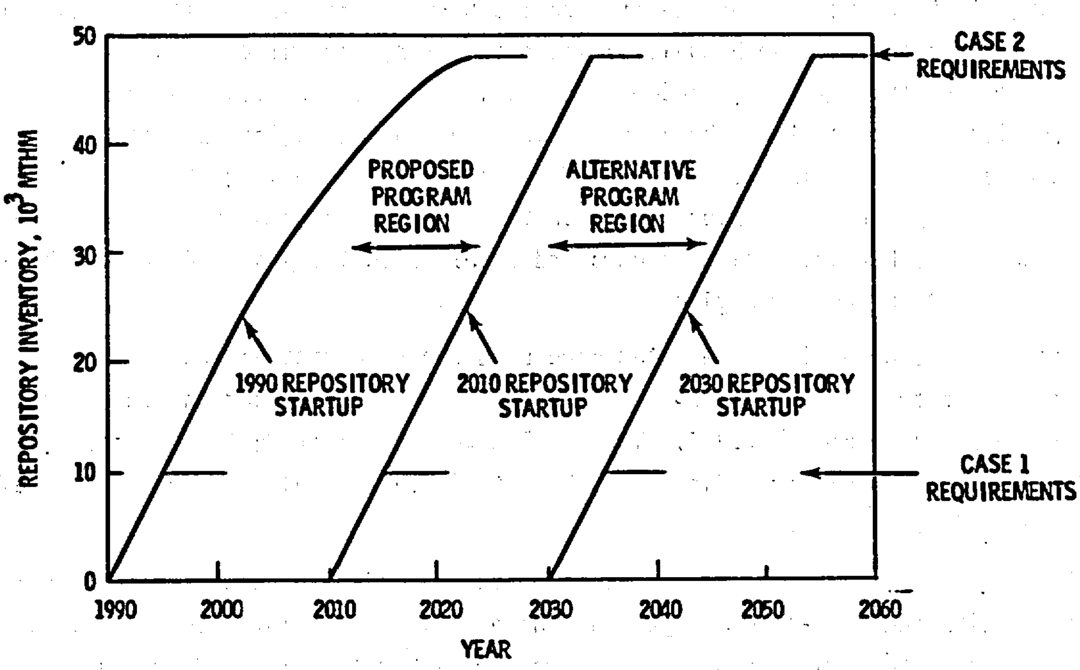

FIGURE 7.3.1. Repository Inventory Accumulations for Cases 1 and 2.

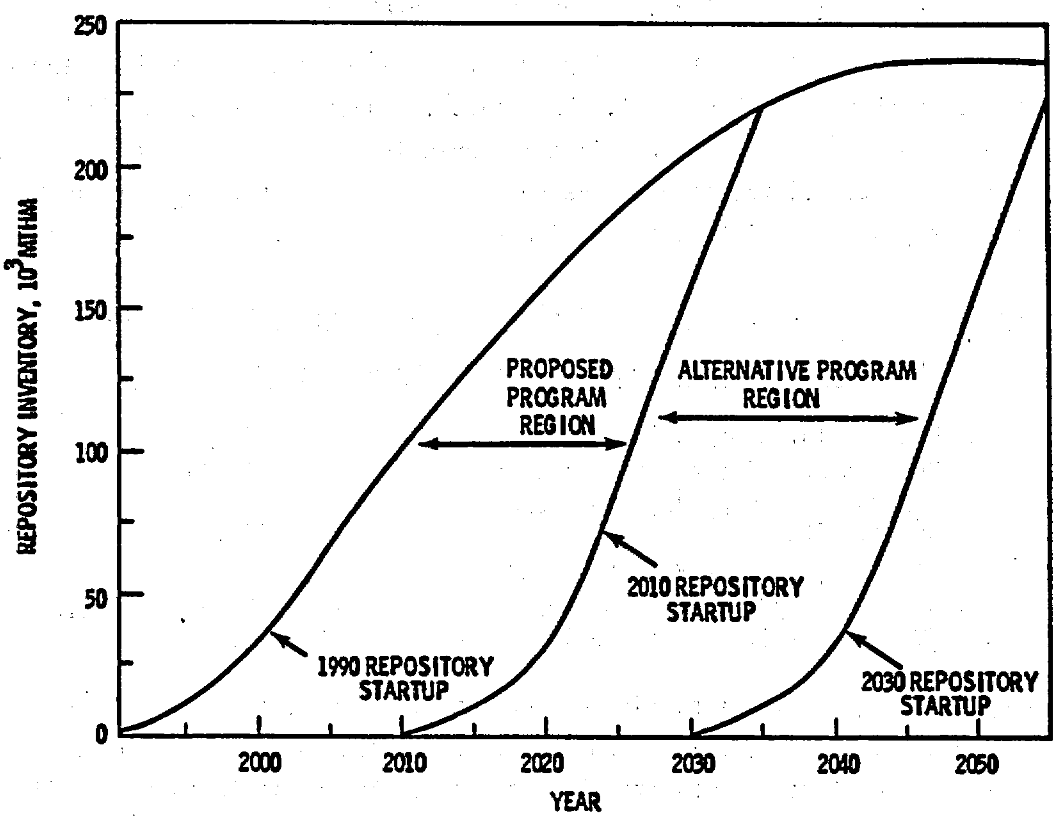

FIGURE 7.3.2. Repository Inventory Accumulation for the Once-Through Cycle in Case 3

receiving rates. The cumulative fuel reprocessed in Case 3 for the two reprocessing startup dates considered is shown in Figure 7.3.3. The repository accumulations of high-level wastes are plotted in Figure 7.3.4. Because of the five-year holdup of high-level waste at the reprocessing plant and because of the differences between the reprocessing rates and the repository receiving capacity, the high-level waste inventory accumulation in the 2010 repository is sensitive to the reprocessing date. For these reasons the region of inventory accumulation representing the proposed program and the region representing the alternative program overlap. The accumulation for the 2010 reprocessing startup and a 2010 repository startup forms the upper bound for the proposed program region while the accumulation for the 1990 reprocessing startup and a 2010 repository startup forms the lower bound for the alternative program region.

For Cases 4 and 5, only mid-range dates were used for reprocessing and repository startup dates. The repository inventory accumulation with the once-through cycle for Cases 4 and 5 are shown in Figure 7.3.5. The cumulative amounts of fuel reprocessed for Cases 4 and 5 are shown in Figure 7.3.6 while the repository accumulations of high-level waste are shown in Figure 7.3.7.

The total number of spent fuel canisters (see Section 4.3.1 for canister descriptions) sent to disposal with the once-through cycle is shown in Table 7.3.1a. Since the total quantity of spent fuel in a given case is the same for either the proposed or the

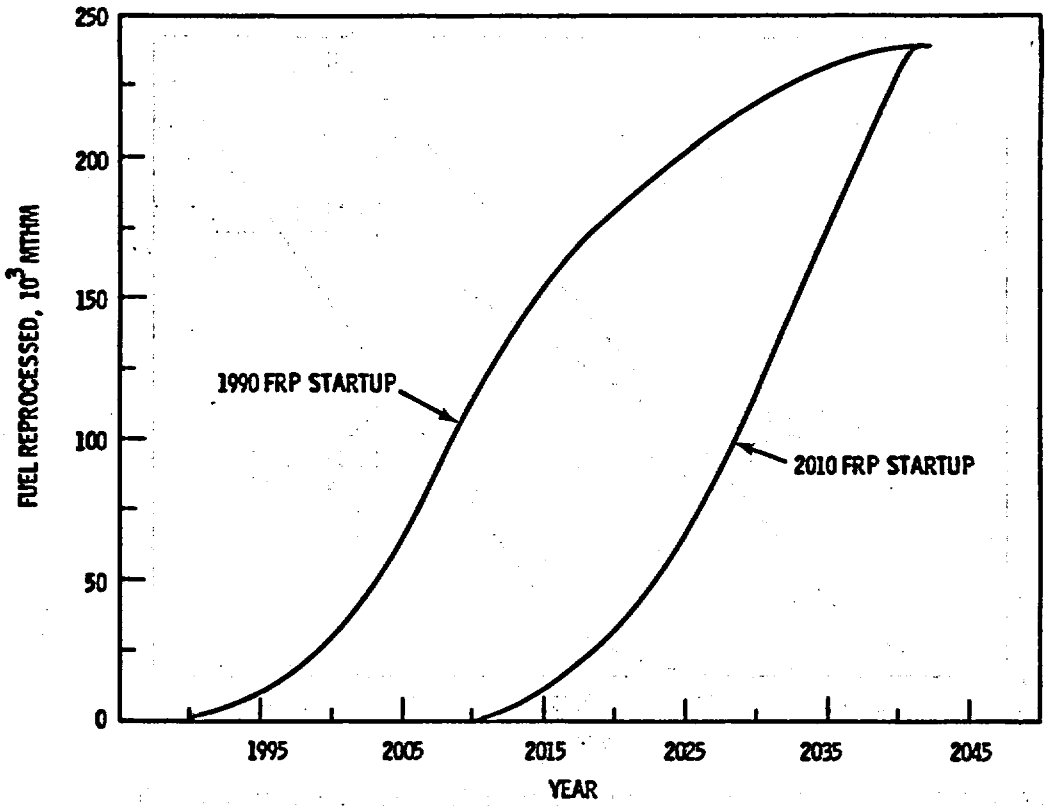

FIGURE 7.3.3. Cumulative Fuel Reprocessed for Case 3

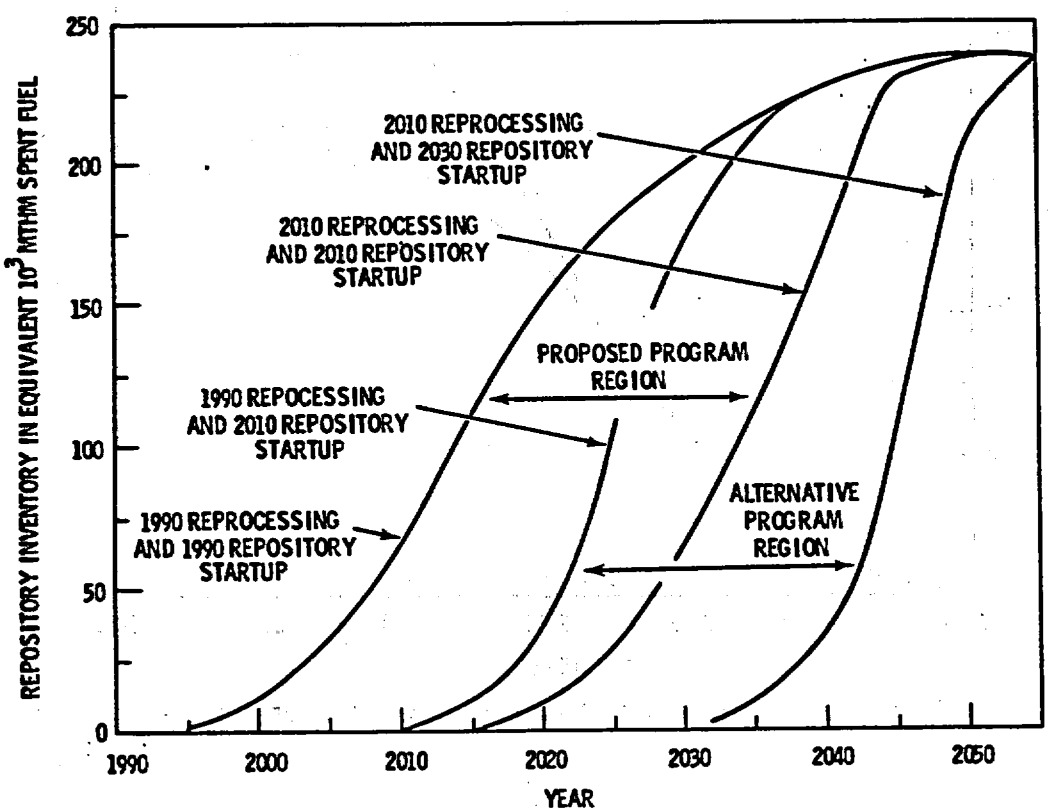

FIGURE 7.3.4. Repository High-Level Waste Inventory Accumulation for Case 3 with Reprocessing

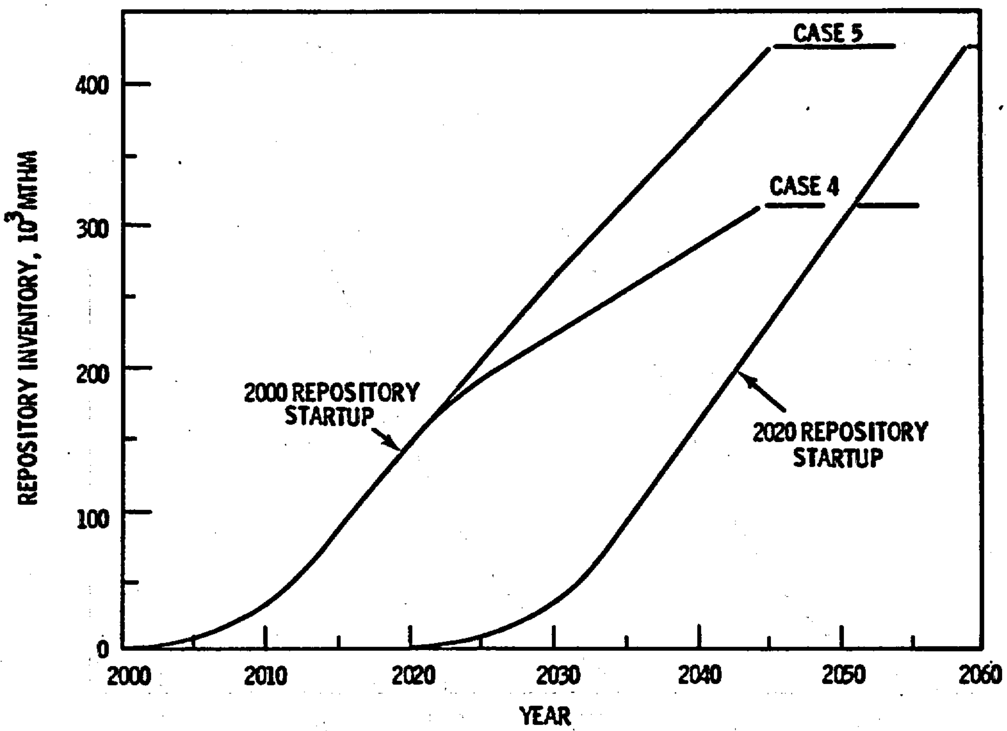

FIGURE 7.3.5. Repository Inventory Accumulation for The Once-Through Cycle in Cases 4 and 5

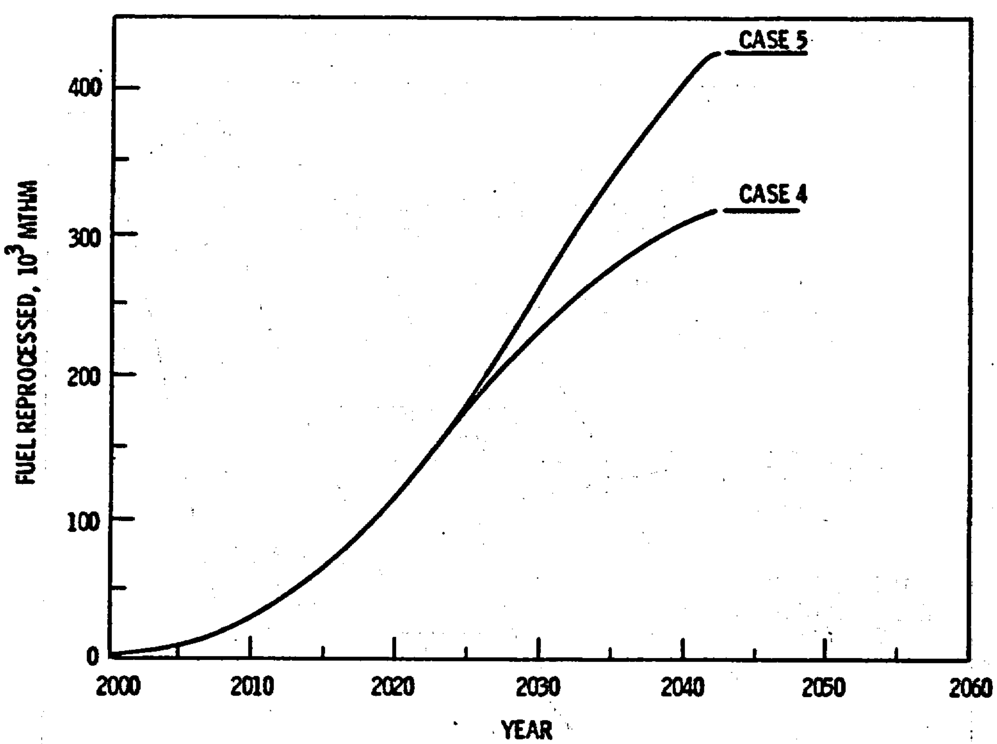

FIGURE 7.3.6. Cumulative Fuel Reprocessed for Cases 4 and 5

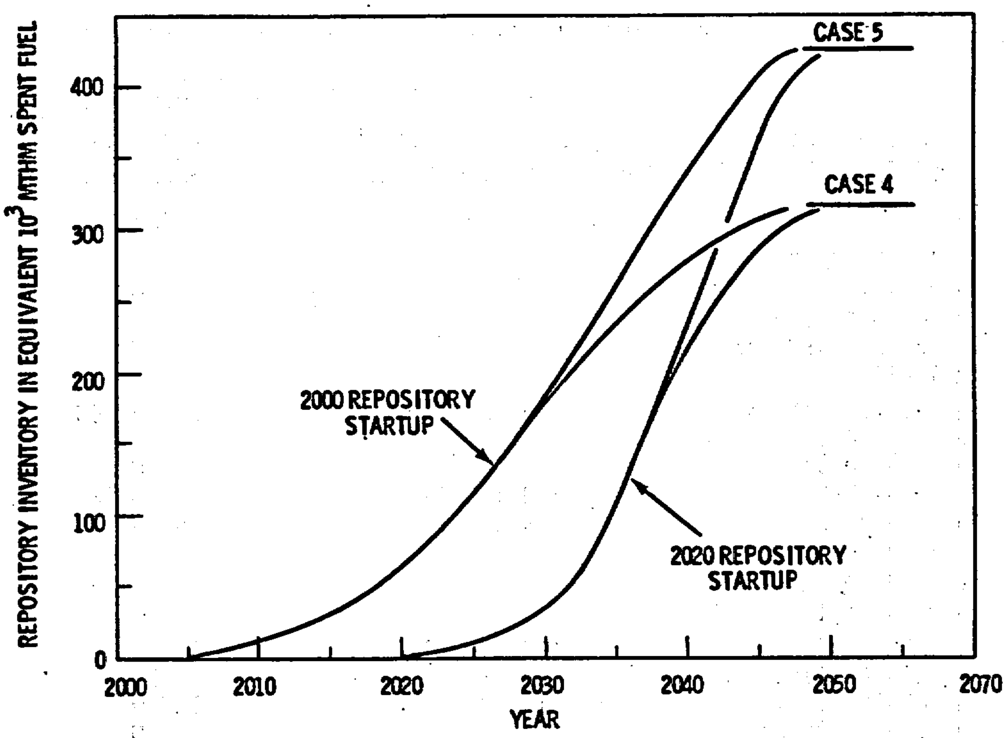

FIGURE 7.3.7. Repository High-Level Waste Inventory Accumulation for Cases 4 and 5 with Reprocessing

TABLE 7.3.1a. Number of Spent Fuel Canisters Sent to Disposal in the Once-Through Cycle

		Thousands of Containers		
Case	Nuclear Power Growth Assumption	Proposed Program (Geologic Disposal Starting 1990 - 2010)	Alternative Program (Disposal Starting 2010 - 2030)	No-Action Alternative
1	Present Inventory Only	35.6	35.6	0
2	Present Capacity and Normal Life	165	165	0
3	250 GWe System by Year 2000 and Normal Life	808	808	0
4	250 GWe system by Year 2000 and Steady State	1,070	1,070	NA(a)
5	500 GWe system by Year 2040	1,440	1,440	NA

(a) NA = not applicable.

alternative program and since we assumed that each fuel assembly would be encapsulated individually for this analysis, the number of canisters is the same for both major alternatives.

The total number of waste containers sent to disposal with the reprocessing cycle is shown in Table 7.3.1b (see Sections 4.3.2 and 4.3.3 for container descriptions). The range of numbers of high-level waste containers results from variations in the allowable heat

TABLE 7.3.1b. Number of Waste Containers Sent to Disposal in Reprocessing Cycle

Case	Nuclear Power Growth Assumption	Thousands of Containers: Proposed Program (Geologic Disposal Starting 1990 - 2010)	Alternative Program (Disposal Starting 2010 - 2030)
1	Present Inventory	NA[a]	NA
2	Present Capacity and Normal Life	NA	NA
3	250 GWe System by Year 2000 and Normal Life		
	• HLW Canisters	80 to 430	80 to 180
	• RH-TRU Canisters	66	66
	• RH-TRU Drums	970	970
	• CH-TRU Drums	530 to 780	530 to 780
	• CH-TRU Boxes	9 to 11	9 to 11
4	250 GWe system by Year 2000 and Steady State		
	• HLW Canisters	140 to 350	114 to 270
	• RH-TRU Canisters	87	87
	• RH-TRU Drums	1,300	1,300
	• CH-TRU Drums	860	860
	• CH TRU Boxes	13	13
5	500 GWe System by Year 2040		
	• HLW Canisters	190 to 530	160 to 390
	• RH-TRU Canisters	117	117
	• RH-TRU Drums	1,740	1,740
	• CH-TRU Drums	1,200	1,200
	• CH-TRU Boxes	19	19

(a) NA = not applicable.

generation rate per canister for the four disposal media and variations in the age, and thus the heat generation rate, of the waste at the time of disposal. The contact-handled TRU waste quantities vary depending on the time reprocessing starts and the quantity of MOX fuel that is reprocessed. See Appendix Table A.1.22 for additional details.

7.3.2 Interim Storage Requirements

The interim storage requirements for spent fuel are controlled in the once-through cycle by the repository receiving capability, and in the reprocessing cycle by the reprocessing capacity. Spent fuel storage requirements in away-from-reactor (AFR) facilities, also referred to as independent spent-fuel storage facilities, are shown in Table 7.3.2 for the once-through cycle and in Table 7.3.3 for the reprocessing cycle. Requirements with or without reprocessing are about the same if repositories start up in the period of 1990 to 2010. However, whereas the storage requirements increase substantially for the once-through cycle with later repositories under the alternative program, the requirements are not changed in the reprocessing case since the storage requirement is controlled by the

TABLE 7.3.2. Comparison of Away-From-Reactor Spent Fuel Storage Requirements for the Program Alternative Using the Once-Through Cycle

		Maximum Storage Requirements, MTHM		
Case	Nuclear Power Growth Assumption	Proposed Program (Geologic Disposal Starting 1990 - 2010)	Alternative Program (Disposal Starting 2010 - 2030)	No Action Alternative
1	Present Inventory Only	0	0	0
2	Present Capacity and Normal Life	7,900 to 30,000	30,000 to 37,000	37,000
3	250 GWe System by Year 2000 and Normal Life	12,000 to 113,000	113,000 to 181,000	197,000
4	250 GWe System by Year 2000 and Steady State	60,000	176,000	NA[a]
5	500 GWe System by Year 2040	61,000	215,000	NA

(a) NA = not applicable.

TABLE 7.3.3. Comparison of Away-From-Reactor Spent Fuel Storage Requirements for the Program Alternative Using the Reprocessing Cycle[a]

		Maximum Storage Requirements, MTHM		
Case	Nuclear Power Growth Assumption	Proposed Program (Geologic Disposal Starting 1990 - 2010)	Alternative Program (Disposal Starting 2010 - 2030)	No Action Alternative
1	Present Inventory Only	NA[b]	NA	NA
2	Present Capacity and Normal Life	NA	NA	NA
3	250 GWe System by Year 2000 and Normal Life	12,000 to 113,000	12,000 to 113,000	NA
4	250 GWe System by Year 2000 and Steady State	62,000	62,000	NA
5	500 GWe System by Year 2040	63,000	63,000	NA

(a) Assumed Reprocessing startup dates range from 1990 to 2010.
(b) NA = Not applicable.

range of reprocessing dates considered. The accumulation and decline of the storage requirements is illustrated for Case 3 in Figures 7.3.8 and 7.3.9 for the once-through cycle and reprocessing cycle, respectively. (See Appendix A.1 for annual requirements of other cases.)

Although in the reprocessing cycle the spent-fuel storage requirements are not increased by delay in repository availability, the storage requirements for the reprocessing

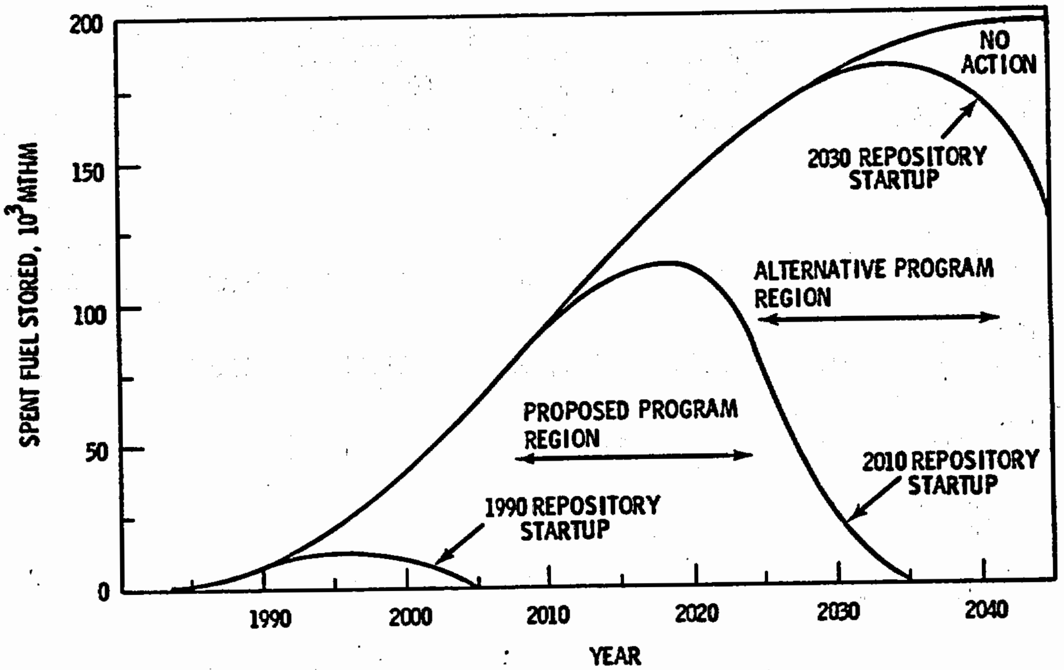

FIGURE 7.3.8. AFR Storage Requirements for Case 3 with the Once-Through Cycle

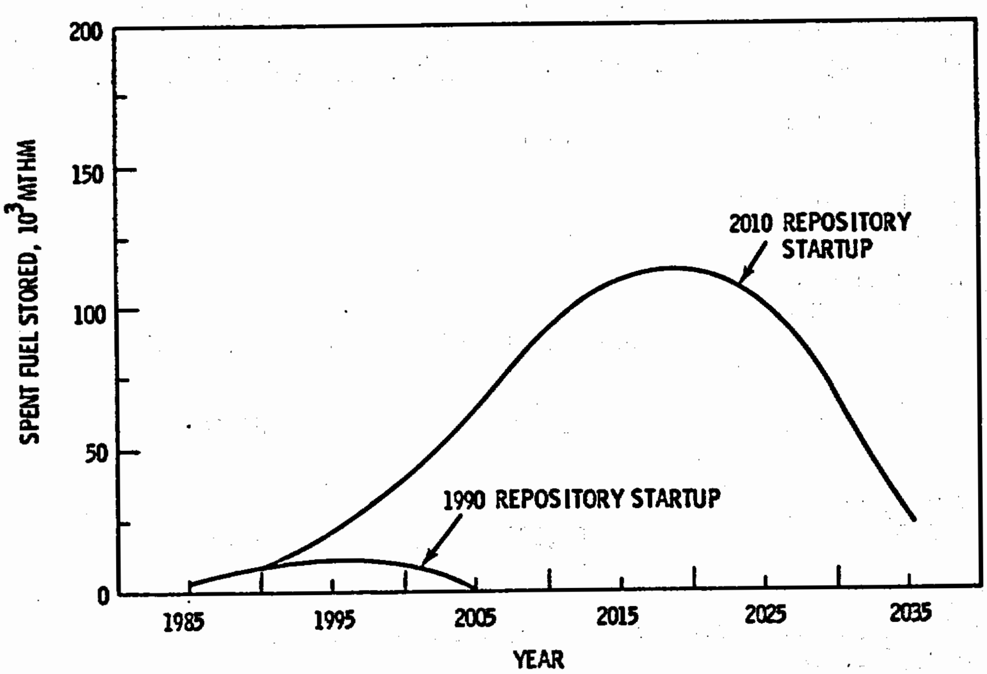

FIGURE 7.3.9. AFR Storage Requirements for Case 3 with Reprocessing

wastes do become substantial for delayed repository availability. This is shown in Table 7.3.4. The range of storage requirements for high-level waste canisters is affected not only by repository availability but also by the heat limitation on canisters for the different geologic media. For example, only about 1/3 as much high-level waste can be placed in a single canister for a repository in shale as can be placed in a canister for a repository in salt (see Section 5.3)..

TABLE 7.3.4. Interim Waste Storage Requirements for the Program Alternatives Using the Reprocessing Cycle[a]

Case	Nuclear Power Growth Assumption	Maximum Number of Containers Stored: Proposed Program (Geologic Disposal Starting 1990 - 2010)	Alternative Program (Disposal Starting 2010 - 2030)	No Action Alternative
1	Present Inventory Only	NA[b]	NA	NA
2	Present Capacity and Normal Life	NA	NA	NA
3	250 GWe System by Year 2000 and Normal Life			
	• HLW Canisters	0 to 85,000[c]	40,000 to 85,000[c]	NA
	• RH-TRU Waste Canisters	0 to 41,000	41,000 to 60,000	NA
	• RH-TRU Waste Drums	0 to 604,000	604,000 to 894,000	NA
	• CH-TRU Waste Drums	0 to 397,000	337,000 to 577,000	NA
	• CH-TRU Waste Boxes	0 to 6,000	6,000 to 9,000	NA
4	250 GWe System by year 2000 and Steady State			
	• HLW Canisters	0	46,000 to 92,000[c]	NA
	• RH-TRU Waste Canisters	0	54,000	NA
	• RH-TRU Waste Drums	0	798,000	NA
	• CH-TRU Waste Drums	0	460,000	NA
	• CH-TRU Waste Boxes	0	8,000	NA
5	500 GWe System by Year 2040			
	• HLW Canisters	0	52,000 to 114,000[c]	NA
	• RH-TRU Waste Canisters	0	63,000	NA
	• RH-TRU Waste Drums	0	936,000	NA
	• CH-TRU Waste Drums	0	599,000	NA
	• CH-TRU Waste Boxes	0	10,000	NA

(a) Assumed reprocessing startup dates range from 1990 to 2010 (see Table 7.1.3).
(b) NA = not applicable.
(c) Range for HLW values for the four disposal media.

For Case 3 under the alternative program, the maximum storage requirements are not as large as one might at first expect considering the time delay to the year 2030 repository startup. This is because of the declining schedule of fuel discharges (see Figure 3.2.3) and the accelerated repository receiving rate used to eliminate the storage backlog (see Figure 7.3.4). For Cases 4 and 5 under the proposed program, the repository starts the same year as reprocessing and there are no interim storage requirements. However, under the alternative program the storage requirements are substantial for these cases.

7.3.3 Transportation Requirements

Transportation requirements are identified here in terms of the number of shipments required. A shipment is defined as one truck cask or one rail or intermodal cask shipment in the case of spent fuel or one truck load or one rail car in the case of reprocessing wastes.

Transportation requirements for the once-through cycle are shown in Table 7.3.5. Truck shipments are the same under the proposed program or the alternative program. This is because it does not matter whether the fuel shipped from the reactor by truck goes to interim storage or the repository. It is only shipped once by truck as shipments from interim storage are assumed to be entirely by rail. Rail shipments can be higher under the alternative program because storage requirements are higher and any fuel shipped to interim storage must be shipped twice--once from the reactor to interim storage and once from interim storage to the repository. Fewer shipments are required under the no-action alternative because some of the fuel remains in the reactor basins and is not shipped at all. Additional details are shown in Appendix A, Table A.7.1.

Transportation requirements for the reprocessing cycle are shown in Table 7.3.6. Transportation requirements range somewhat higher under the alternative program than under the proposed program because more shipments are required to interim storage as a result of

TABLE 7.3.5. Comparison of Transportation Requirements for the Program Alternative Using the Once-Through Fuel Cycle

			Number of Spent Fuel Shipments		
Case	Nuclear Power Growth Assumption	Transport Mode	Proposed Program (Geologic Disposal Starting 1990 - 2010)	Alternative Program (Disposal Starting 2010 - 2030)	No Action Alternative
1	Present Inventory Only	Rail Truck	2,300 2,300	2,300 2,300	0 0
2	Present Capacity Normal Life	Rail Truck	13,300 to 18,000 11,000	18,000 to 19,000 11,000	8,400 8,600
3	250 GWe by Year 2000 and Steady State	Rail Truck	61,000 to 89,000 56,000	89,000 to 96,000 56,000	45,000 46,000
4	250 GWe System by Year 2000 and Steady State	Rail Truck	97,000 73,000	127,000 73,000	NA[(a)] NA
5	500 GWe by Year 2040	Rail Truck	126,000 99,000	170,000 99,000	NA NA

(a) NA = not applicable.

TABLE 7.3.6. Comparison of Total Transportation Requirements for the Program Alternative Using the Reprocessing Fuel Cycle[a]

Case	Nuclear Power Growth Assumption	Transport Mode	Number of Shipments: Proposed Program (Geologic Disposal Starting 1990 - 2010)	Number of Shipments: Alternative Program (Disposal Starting 2010 - 2030)
1	Present Inventory Only	NA[b]	NA	NA
2	Present Capacity and Normal Life	NA	NA	NA
3	250 GWe System by Year 2000 and Normal Life	Rail	90,000 to 119,000	117,000 to 147,000
		Truck	182,000 to 314,000	182,000 to 317,000
4	250 GWe System by Year 2000 and Steady State	Rail	136,000	176,000
		Truck	250,000	412,000
5	500 GWe System by Year 2040	Rail	179,000	233,000
		Truck	343,000	566,000

(a) Assumed reprocessing startup dates range from 1990 to 2010; (see Table 7.1.3)
(b) NA = not applicable.

the potentially greater delay in repository availability. Requirements for truck shipments are much larger than in the once-through cycle because of the assumption that all TRU waste drums and boxes are shipped by truck. These wastes could be shipped by rail; in that case, only 1/2 to 1/3 as many shipments would be required. More details of the transportation requirements with the reprocessing cycle are shown in Appendix A, Table A.7.2.

7.3.4 Age of the Waste at Disposal

A potentially beneficial aspect of delayed repository availability under the alternative program is the aging of the waste, which reduces radioactivity and heat generation rates. The maximum and minimum ages at disposal for spent fuel from the once-through cycle and high-level waste from the reprocessing cycle are shown in Tables 7.3.7 and 7.3.8, respectively. To illustrate this aspect more fully, the ages of spent fuel and high-level waste for Case 3 are plotted as a function of time in Figures 7.3.10 and 7.3.11 for the once-through and the reprocessing cycles.

The lower thermal output for the aged waste would permit either more waste to be placed in individual canisters and a higher areal loading of the repositories, or could be used to provide a greater level of technical conservatism by allowing reduced temperatures for emplaced wastes. For this analysis, the quantity of high-level waste placed in individual canisters has been adjusted to take advantage of the lower thermal output of the aged waste, and the calculated repository requirements take into account the lower thermal output of the aged waste. The relationship between age of the waste and repository capacity is discussed in Section 5.3.3 and Appendix K.

TABLE 7.3.7. Maximum (and Minimum) Age of Spent Fuel Entering the Repository Using the Once-Through Cycle, Years

Case	Nuclear Power Growth Assumption	Proposed Program (Geologic Disposal Starting 1990 - 2010)	Alternative Program (Disposal Starting 2010 - 2030)
1	Present Inventory Only	18(14) to 38(34)	38(34) to 58(54)
2	Present Capacity and Normal Life	18(5) to 38(18)	38(18) to 58(38)
3	250 GWe System by Year 2000 and Normal Life	18(5) to 38(5)	38(5) to 58(19)
4	250 GWe System by Year 2000 and Steady State	28(5)	48(12)
5	500 GWe System by Year 2040	28(5)	48(20)

TABLE 7.3.8. Maximum (and minimum) Age of High-Level Waste Entering the Repository using the Reprocessing Cycle,[a] Years[b]

Case	Nuclear Power Growth Assumption	Proposed Program (Geologic Disposal Starting 1990 - 2010)	Alternative Program (Disposal Starting 2010 - 2030)
1	Present Inventory Only	NA[c]	NA
2	Present Capacity and Normal Life	NA	NA
3	250 GWe System by Year 2000 and Normal Life	23(6.5) to 43(7)	38(6.5) to 58(13)
4	250 GWe System by Year 2000 and Steady State	33(6.5)	48(8)
5	500 GWe System by year 2040	33(6.5)	48(8)

(a) Assumes reprocessing startup dates range from 1990 to 2010 (see Table 7.1.3).
(b) Years from reactor discharge.
(c) NA = not applicable.

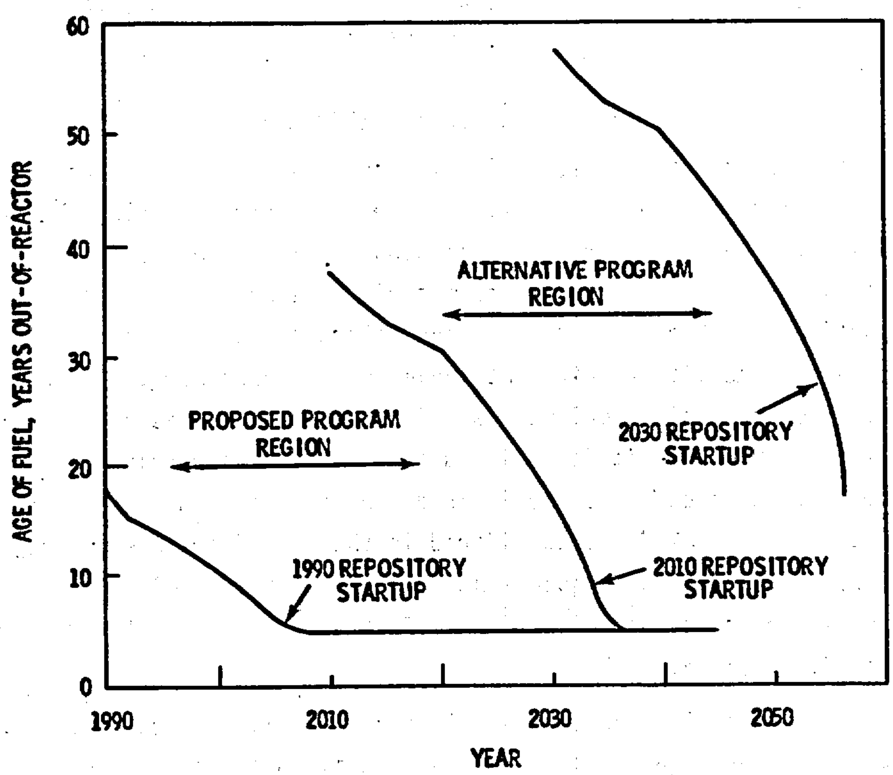

FIGURE 7.3.10. Age of Fuel Entering Repository for Case 3 with the Once-Through Cycle

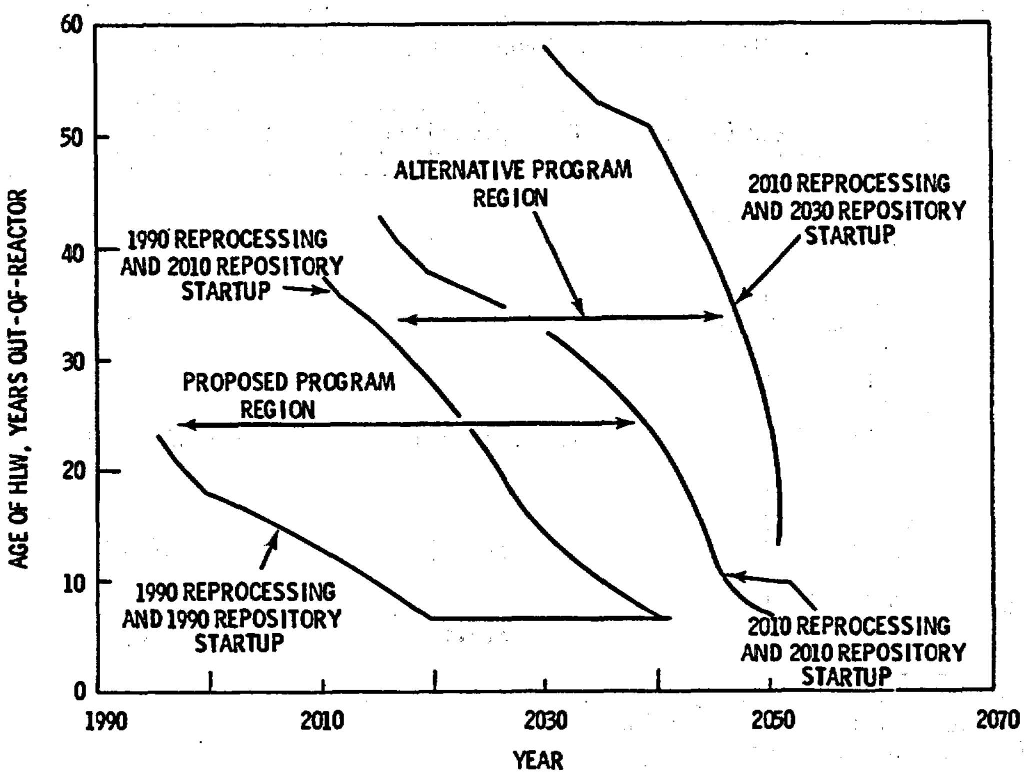

FIGURE 7.3.11. Age of High-Level Waste Entering Repository for Case 3 with the Reprocessing Cycle

7.3.5 Facility Requirements

To estimate resource requirements, it is first necessary to define the number of waste management facilities required in each case. In the once-through cycle, the only facilities required in addition to the repository and packaging facility are the independent fuel storage facilities for interim storage of the spent fuel. The number of these facilities required is proportional to the maximum spent fuel storage requirements shown in Table 7.3.2; a separate facility requirement table is not shown here. A 3,000 MTHM independent spent-fuel storage basin model was used in this Statement as a basis for resource requirement estimates. However, it is believed that facilities ranging up to 20,000 MTHM capacity might be used in cases where the interim storage requirements are very large. (Storage facilities up to 18,000 MTHM are considered in the U.S. Spent Fuel Policy Statement (DOE/EIS-0015 1980). The resource requirements and costs would decline somewhat as individual facility sizes increase because of scaling-effect efficiencies but radiation total releases would not be affected.

For the reprocessing cycle, the spent-fuel storage facility requirements would be proportional to the maximum storage requirement shown in Table 7.3.3. Other waste management facility requirements would be proportional to the number of fuel reprocessing plants and MOX fuel-fabrication plants utilized to process and recycle the spent fuel. Requirements for these facilities are shown in Table 7.3.9.

The number of equivalent 30-year-life plants utilized through the year 2040 was used to estimate resource requirements rather than number of plants started up. (Average utilization or capacity factor for a reprocessing plant was assumed to be 80% of on-stream design capacity and for a MOX fuel-fabrication plant a 65% factor was assumed.) It was assumed that the balance of the facilities started up would be utilized for continuing requirements outside the boundaries of the systems studied here. Both the number of startups and equivalent 30-year-life plants are shown in Table 7.3.9.

The number of repositories required is sensitive to the geologic medium. In the case of spent fuel, for example, the criteria utilized in this Statement indicate that the underground area required to store wastes in salt or shale is approximately twice that needed to store wastes in granite or basalt. For the reprocessing cycle wastes, salt compares favorably with granite and basalt, but shale requires on the order of twice the area required for the other three media examined. Taking into account the range of requirements for the four media considered here, Table 7.3.10 shows the range of 800-hectare (2,000-acre) repositories required for both the once-through and the reprocessing cycles. Further details can be found in Appendix Tables A.10.1 and A.10.2.

Although the range of requirements shown in Table 7.3.10 results largely from the range of geologic media considered, the range is also affected by the age of the waste. An older waste generates less heat and, as a consequence, permits somewhat more efficient use of repository space. The effect of waste age on repository capacity is discussed in Section 5.3.3.

Since significant improvements may yet be possible in both the once-through cycle repository concept and the reprocessing cycle repository concept, conclusions regarding relative repository requirements by fuel cycle should be considered as preliminary. The generally larger repository requirement for reprocessing wastes (salt is an exception) results from the additional placement area required for TRU wastes. (An illustration of the relative repository area requirements for each waste type can be found in DOE/ET-0028, Vol. 4, Tables 7.4.2 and 7.5.3.)

TABLE 7.3.9. Fuel Reprocessing and MOX Fuel Fabrication Plant Requirements

Case	Nuclear Power Growth Assumption	2000 MTHM Fuel Reprocessing Plants		400 MTHM MOX Fuel Fabrication Plants	
		Startups	Equivalent 30-yr-life plant Utilized	Startups	Equivalent 30-yr-life-Plants Utilized
1	Present Inventory Only	NA[a]	NA	NA	NA
2	Present Capacity and Normal Life	NA	NA	NA	NA
3	250 GWe System by Year 2000 and Normal Life	6	4	4 to 7	1.8 to 5.6
4	250 GWe System by year 2000 and Steady State	6	5.1	8	6.6
5	500 GWe System by year 2040	9	6.8	15	10.4

(a) NA = Not Applicable.

TABLE 7.3.10. Number of 800-hectare[(a)] Repositories Required

Case	Nuclear Power Growth Assumption	Once-Through Cycle	Reprocessing Cycle
1	Present Inventory Only	0.03 to 0.1	NA
2	Present Capacity and Normal Life	0.2 to 0.7	NA
3	250 GWe System by Year 2000 and Normal Life	1 to 4	2 to 5
4	250 GWe System by Year 2000 and Steady State	2 to 5	3 to 6
5	500 GWe System by Year 2000	2 to 7	4 to 9

(a) 800 hectares = 2000 acres.
(b) NA = not applicable.

7.3.6 Equilibrium Requirements for Equilibrium Steady-State Systems

One of the purposes for Case 4 was to illustrate the level of continuing requirements in a steady-state nuclear system--in this case, 250 GWe. Table 7.3.11 shows these equilibrium requirements in terms of spent fuel disposal or reprocessing requirements, annual waste shipments and the number of years to fill an 800-hectare repository in the four geologic media. Requirements for other sizes of steady-state systems will be directly proportional to these requirements. For example, a 500 GWe steady-state system would have twice the requirements shown in Table 7.3.11. Data are provided on the number of years to fill repositories for waste ages of 5 and 50 years to lend perspective on the age variable. A significant improvement for the 50-year-old waste is indicated in all media for the reprocessing wastes and for spent fuel in granite or basalt, but relatively small improvements are shown for spent fuel in salt and shale.

TABLE 7.3.11. Equilibrium Requirements for Case 4 (250 GWe Steady State)

	Spent Fuel to Disposal or Reprocessing, MTHM	Annual Waste Shipments[(a)]		Time Required to Fill an 800-hectare Repository			
		Truck	Rail	5-yr-old Spent Fuel	50-yr-old Spent Fuel	5-yr-old HLW	50-yr-old HLW
One-Through Cycle							
Spent Fuel	6000	1400	1400				
Salt				10	11		
Granite				24	32		
Shale				12	15		
Basalt				24	32		
Reprocessing Cycle							
Spent Fuel	6000	1400	1400				
HLW and Other Wastes		3400	810				
Salt						11	21
Granite						11	23
Shale						6	11
Basalt						11	20

(a) A shipment is defined as one rail car or one truck load.

7.3.7 Plutonium Disposition

Examination of the disposition of plutonium helps to explain differences in the composition of the waste produced in the different nuclear growth cases and the effect that the reprocessing date has on the waste compositions (The reprocessing date effects the amount of recycle achieved within the time frame of this analysis.) Table 7.3.12 shows the plutonium disposition in both the once-through cycle and the reprocessing cycle. Disposition in the once-through cycle is straightforward--all of the plutonium goes to the repository with the spent fuel. With the reprocessing cycle, the situation is more complex. Much of the plutonium that is recycled is eliminated by fissioning. However, recycle of plutonium in mixed plutonium and uranium oxide fuel also produces more plutonium by conversion of ^{238}U. Thus, the total amount of plutonium generated in the reprocessing cycle is always larger than the total amount of plutonium in the once-through cycle spent fuel. Approximately 99% of the plutonium in the spent fuel is recovered by reprocessing and (excluding third-recycle discard) a little more than one percent of the plutonium ends up in the wastes; approximately 0.5% is in the high-level waste and the balance is dispersed in the TRU wastes. Plutonium recycle also produces more higher atomic number actinides (e.g., americium, neptunium and curium), which also end up in the waste.

At the end of the reactor operation period in each reprocessing case, there is some plutonium remaining in the fuel as well as plutonium in the reprocessing pipeline. This plutonium is shown in Table 7.3.12 as plutonium not recycled. It is assumed to be recovered by reprocessing but is not recycled in this system. We assume that other reactors that continue to operate outside of this system would, except for third-recycle plutonium, utilize this plutonium. Thus, except for the third-recycle portion, the plutonium not recycled is not considered for disposal in this Statement. Presumably, there will come a time when the industry will be shut down and the excess plutonium at that time will require disposal. However, before that time, steps could be taken to minimize the amount of plutonium left in the pipeline. With proper planning, the amount of plutonium requiring disposal could be reduced to the plutonium contained in the last batches of spent fuel. Since there would be no incentive for further reprocessing at that time, this spent fuel could be disposed of as spent fuel in the same manner as in the once-through cycle.

We assume here that the plutonium recovered from the third recycle is not recycled and that it is discarded in the high-level waste. Table 7.3.12 shows this to be a relatively small amount. In a real system, whether or not this plutonium is recycled will be primarily an economic determination. Recycle could be continued until all of the plutonium is either fissioned or transmuted to higher actinides, which are then is discarded in the waste.

The two reprocessing dates used for Case 3 illustrate how sensitive the plutonium disposition is to reprocessing dates. Less than one-third as much plutonium is recycled when reprocessing starts in 2010 as when reprocessing starts in 1990. This is because of: 1) the large inventory of spent fuel accumulated when reprocessing starts, 2) a preference given to first-recycle plutonium relative to second- or third-recycle plutonium because of its higher fuel value, 3) the limitation on recycle MOX fuel to 50% of the equilibrium reload

TABLE 7.3.12. Plutonium Disposition Within the Timeframe of the Analysis

Case	Nuclear Power Growth Assumption	Once-Through Cycle Total Pu in Spent Fuel, MT	Reprocessing Cycle				
			Year Reprocessing Starts	Total Pu Generated, MT	Pu Recycled, MT	Pu Not Recycled, MT	Third Recycle Discard, MT
1	Present Inventory Only	36	NA[a]	NA	NA	NA	NA
2	Present Capacity and Normal Life	375	NA	NA	NA	NA	NA
3	250 GWe System by Year 2000 and Normal life	1,898	1990	3,429	2,308	1,121	152
			2010	2,160	644	1,516	0
4	250 GWe System by Year 2000 and Steady State	2,225	2000	3,779	2,146	1,633	0
5	500 GWe System by Year 2040	2,911	2000	5,147	3,584	1,559	0

(a) NA = not applicable.

fuel, 4) the long time for spent recycle fuel to work its way through the inventory to reprocessing, and 5) the year 2040 cutoff date for this analysis. No third-recycle fuel is irradiated in the Year 2010 reprocessing case. The same effect is noted in Case 4 and Case 5. We calculate that at equilibrium, 4 MT of third-recycle plutonium would be discharged for each 1,000 MT of equilibrium plus recycle reload fuel charged (equilibrium reload fuel accounts for approximately 80% of the total fuel). Thus, for Case 3 for example, where 239,000 MT of fuel are charged, the eventual implied commitment for third-recycle plutonium disposal is approximately 780 MT.

7.3.8 Radioactivity Inventory in Disposal Repositories

The total radioactivity and the total heat output from the entire inventory of all wastes sent to disposal from the entire system are summarized in Tables 7.3.13 through 7.3.16. These tables show the activity and heat output from year 2070 at periodic intervals for the next 1 million years for each of the nuclear growth cases. By the year 2070, all wastes have been placed in the repositories and much of the shorter life activities have decayed to low levels. Detailed tables showing the breakdown of radioactivity and heat output by individual nuclides are included in Appendix A.2 and A.3.

Table 7.3.13 shows the radioactivity inventory for all the fission and activation products. The radioactivity here is roughly proportional to the total energy produced in each case (see Table 7.1.1). The fission and activation product inventory for the reprocessing cases is closely similar to the fission and activation product inventory for the once-through cases.

Table 7.3.14 summarizes the total radioactivity inventory for all of the actinides and their daughter nuclides. The activity inventories in the once-through cases are roughly proportional to the energy generated in each case. This is also true for the reprocessing cases. However, the actinide inventories for comparable reprocessing and once-through cases are substantially different. The actinide activity initially is much higher with the once-through cycle wastes. This is because these wastes contain all of the plutonium present in the spent fuel. However, the recycle wastes contain a much higher level of the higher actinides--americum, curium, etc. Thus, the difference in total actinide activity inventories is not as large as one might expect based just on the plutonium content, and the differences become smaller in later years. Reprocessing Case 3 shows that the reprocessing date significantly effects the total actinide activity inventory in the wastes.

Table 7.3.15 shows total heat output for the fission and activation products and Table 7.3.16 shows heat output for the actinides and their daughter nuclides. These tables show that in all cases, the heat output is dominated by the actinides after the first 500 years.

Comparisons of the toxicity of radioactive wastes on the basis of hazard indices is discussed in Section 3.4. The relative toxicities of the once-through cycle and

TABLE 7.3.13. Total Radioactivity Inventory of All Fission and Activation Products in All Repositories(a)

Fuel Cycle	Case	Reprocessing Date	Curies: Year 2070	500 Years	1000 Years	5000 Years	10,000 Years	50,000 Years	100,000 Years	500,000 Years	1,000,000 Years
Once-Through	1	NA(b)	2.90×10^{8}	4.56×10^{5}	1.71×10^{5}	1.61×10^{5}	1.57×10^{5}	1.33×10^{5}	1.12×10^{5}	4.18×10^{4}	2.18×10^{4}
	2	NA	2.66×10^{9}	3.01×10^{6}	1.07×10^{6}	1.01×10^{6}	9.79×10^{5}	8.34×10^{5}	7.02×10^{5}	2.62×10^{5}	1.37×10^{5}
	3	NA	1.85×10^{10}	1.65×10^{7}	5.42×10^{6}	5.07×10^{6}	4.92×10^{6}	4.19×10^{6}	3.52×10^{6}	1.31×10^{6}	6.85×10^{5}
	4	NA	2.82×10^{10}	2.26×10^{7}	7.13×10^{6}	6.66×10^{6}	6.47×10^{6}	5.50×10^{6}	4.63×10^{6}	1.72×10^{6}	8.99×10^{5}
	5	NA	4.16×10^{10}	3.14×10^{7}	9.70×10^{6}	9.05×10^{6}	8.79×10^{6}	7.48×10^{6}	6.29×10^{6}	2.34×10^{6}	1.22×10^{6}
Reprocessing	3	1990	1.75×10^{10}	1.64×10^{7}	5.35×10^{6}	5.01×10^{6}	4.87×10^{6}	4.16×10^{6}	3.51×10^{6}	1.32×10^{6}	6.98×10^{5}
		2010	1.80×10^{10}	1.64×10^{7}	5.42×10^{6}	5.07×10^{6}	4.92×10^{6}	4.19×10^{6}	3.53×10^{6}	1.33×10^{6}	7.05×10^{5}
	4	2000	2.69×10^{10}	2.24×10^{7}	7.08×10^{6}	6.61×10^{6}	6.43×10^{6}	5.48×10^{6}	4.62×10^{6}	1.74×10^{6}	9.20×10^{5}
	5	2000	3.95×10^{10}	3.12×10^{7}	9.62×10^{6}	8.98×10^{6}	8.72×10^{6}	7.44×10^{6}	6.27×10^{6}	2.36×10^{6}	1.25×10^{6}

(a) Beyond 2070, time intervals are measured from 1980.
(b) NA = not applicable.

TABLE 7.3.14. Total Radioactivity Inventory of All Actinide and Daughter Nuclides in All Repositories(a)

Fuel Cycle	Case	Reprocessing Date	Curies: Year 2070	500 Years	1000 Years	5000 Years	10,000 Years	50,000 Years	100,000 Years	500,000 Years	1,000,000 Years
Once-Through	1	NA(b)	5.01×10^{7}	2.01×10^{7}	1.22×10^{7}	4.75×10^{6}	3.51×10^{6}	7.98×10^{5}	3.05×10^{5}	1.64×10^{5}	1.29×10^{5}
	2	NA	4.33×10^{8}	1.26×10^{8}	7.38×10^{7}	2.61×10^{7}	1.91×10^{7}	4.20×10^{6}	1.69×10^{6}	9.75×10^{5}	7.52×10^{5}
	3	NA	3.06×10^{9}	6.43×10^{8}	3.75×10^{8}	1.31×10^{8}	9.56×10^{7}	2.11×10^{7}	8.49×10^{6}	4.89×10^{6}	3.77×10^{6}
	4	NA	4.90×10^{9}	8.55×10^{8}	4.97×10^{8}	1.73×10^{8}	1.26×10^{8}	2.78×10^{7}	1.12×10^{7}	6.43×10^{6}	4.97×10^{6}
	5	NA	7.38×10^{9}	1.17×10^{9}	6.79×10^{8}	2.35×10^{8}	1.72×10^{8}	3.78×10^{7}	1.52×10^{7}	8.72×10^{6}	6.75×10^{6}
Reprocessing	3	1990	1.43×10^{9}	2.90×10^{8}	1.53×10^{8}	3.70×10^{7}	2.54×10^{7}	4.05×10^{6}	2.16×10^{6}	2.31×10^{6}	2.01×10^{6}
		2010	8.22×10^{8}	3.49×10^{8}	1.63×10^{8}	1.10×10^{7}	7.66×10^{6}	1.72×10^{6}	1.33×10^{6}	2.06×10^{6}	1.93×10^{6}
	4	2000	1.18×10^{9}	3.55×10^{8}	1.75×10^{8}	2.57×10^{7}	1.78×10^{7}	3.14×10^{6}	1.97×10^{6}	2.72×10^{6}	2.53×10^{6}
	5	2000	1.85×10^{9}	4.99×10^{8}	2.48×10^{8}	3.99×10^{7}	2.75×10^{7}	4.60×10^{6}	2.69×10^{6}	3.56×10^{6}	3.30×10^{6}

(a) Beyond 2070, time intervals are measured from 1980.
(b) NA = not applicable.

TABLE 7.3.15. Heat Output of Total Inventory of all Fission and Activation Products in All Repositories(a)

Fuel Cycle	Case	Reprocessing Date	Watts								
			Year 2070	500 Years	1000 Years	5000 Years	10,000 Years	50,000 Years	100,000 Years	500,000 Years	1,000,000 Years
Once-Through	1	NA(b)	8.85×10^5	6.79×10^2	2.95×10^2	2.83×10^2	2.76×10^2	2.29×10^2	1.84×10^2	4.13×10^1	1.09×10^1
	2	NA	8.18×10^6	4.53×10^3	1.87×10^3	1.79×10^3	1.75×10^3	1.45×10^3	1.16×10^3	2.59×10^2	6.84×10^1
	3	NA	5.72×10^7	2.50×10^4	9.41×10^3	8.99×10^3	8.77×10^3	7.28×10^3	5.82×10^3	1.30×10^3	3.43×10^2
	4	NA	8.73×10^7	3.43×10^4	1.24×10^4	1.18×10^4	1.15×10^4	9.56×10^3	7.65×10^3	1.71×10^3	4.51×10^2
	5	NA	1.29×10^8	4.80×10^4	1.68×10^4	1.61×10^4	1.57×10^4	1.30×10^4	1.04×10^4	2.32×10^3	6.12×10^2
Reprocessing	3	1990	5.38×10^7	2.60×10^4	9.98×10^3	9.53×10^3	9.29×10^3	7.67×10^3	6.10×10^3	1.31×10^3	3.48×10^2
		2010	5.55×10^7	2.51×10^4	9.53×10^3	9.10×10^4	8.87×10^3	7.35×10^3	5.87×10^3	1.30×10^3	3.47×10^2
	4	2000	8.31×10^7	3.53×10^4	1.29×10^4	1.23×10^4	1.20×10^4	9.91×10^3	7.90×10^3	1.72×10^3	4.56×10^2
	5	2000	1.22×10^8	4.94×10^4	1.76×10^4	1.68×10^4	1.64×10^4	1.35×10^4	1.08×10^4	2.34×10^3	6.20×10^2

(a) Beyond 2070, time intervals are measured from 1980.
(b) NA = not applicable.

TABLE 7.3.16. Heat Output of Total Inventory of All Actinide and Daughter Nuclides in All Repositories(a)

Fuel Cycle	Case	Reprocessing Date	Watts								
			Year 2070	500 Years	1000 Years	5000 Years	10,000 Years	50,000 Years	100,000 Years	500,000 Years	1,000,000 Years
Once-Through	1	NA(b)	1.26×10^6	6.53×10^5	3.91×10^5	1.46×10^5	1.08×10^5	2.38×10^4	8.21×10^3	3.69×10^3	2.90×10^3
	2	NA	8.53×10^6	4.10×10^6	2.37×10^6	7.98×10^5	5.83×10^5	1.25×10^5	4.51×10^4	2.22×10^4	1.71×10^4
	3	NA	4.40×10^7	2.10×10^7	1.20×10^7	4.00×10^6	2.92×10^6	6.25×10^5	2.26×10^5	1.12×10^5	8.59×10^4
	4	NA	5.84×10^7	2.79×10^7	1.60×10^7	5.29×10^6	3.87×10^6	8.25×10^5	2.98×10^5	1.47×10^5	1.13×10^5
	5	NA	7.99×10^7	3.81×10^7	2.18×10^7	7.20×10^6	5.26×10^6	1.12×10^6	4.04×10^5	1.99×10^5	1.54×10^5
Reprocessing	3	1990	3.15×10^7	9.06×10^6	4.60×10^6	8.98×10^5	6.26×10^5	1.11×10^5	5.36×10^4	5.46×10^4	4.79×10^4
		2010	2.63×10^7	1.14×10^7	5.28×10^6	2.45×10^5	1.73×10^5	4.29×10^4	3.05×10^4	4.84×10^4	4.58×10^4
	4	2000	3.66×10^7	1.13×10^7	5.43×10^6	5.75×10^5	4.06×10^5	8.15×10^4	4.65×10^4	6.41×10^4	6.01×10^4
	5	2000	5.60×10^7	1.58×10^7	7.65×10^6	8.94×10^5	6.30×10^5	1.21×10^5	6.43×10^4	8.37×10^4	7.83×10^4

(a) Beyond 2070, time intervals are measured from 1980.
(b) NA = not applicable.

reprocessing cycle wastes are compared in Table 7.3.17. The index employed here is the amount of water required to dilute one MTHM equivalent of the waste to drinking water standards (10 CFR 20) divided by the amount of water (8.7×10^7 m^3) required to dilute the original uranium ore to drinking water standards.[a] An index of 1.0 means the toxicity hazard is equivalent to the original uranium ore. Detailed tables summing the dilution hazard-index for all of the significant fission and activation products and the actinides and their decay products are presented in Appendix A.4

The data in Table 7.3.17 show essentially equivalent relative hazard indices for all of the once-through cycle cases. Equivalence (index = 1) with uranium ore is reached after about 10,000 years.

Except at the beginning where they are closely similar, the reprocessing waste indices are somewhat lower than the once-through indices and reflect sensitivity to the amount of plutonium recycle achieved as identified by the reprocessing date. Equivalence with uranium ore is reached between 1000 and 2000 years after repository closure.

Nuclides that account for 90-plus percent of the hazard index are listed in Table 7.3.18 for several time periods. Only Case 3 is shown for the once-through cycle since all once-through cases are similar.

Initially, in both cycles, ^{90}Sr accounts for 95+% of the hazard index. At 1000 years the principal contributors in the once-through cycle are ^{241}Am, ^{240}Pu and ^{239}Pu and in the reprocessing cycle are ^{241}Am, ^{243}Am and ^{240}Pu. At 10,000 years the principal contributors in the once-through cycle are ^{239}Pu and ^{240}Pu, while in the reprocessing cycle they are ^{243}Am, ^{240}Pu and ^{239}Pu. For the 100,000- to 1,000,000-year period in the once-through cycle, ^{226}Ra and ^{210}Pb (both daughters of ^{238}U) are the principle hazards, while in the reprocessing cycle, the principle contributors include ^{229}Th, ^{129}I, and ^{237}Np in addition to ^{226}Ra.

It should be noted that although this index is one way to measure relative toxicity of the wastes it says nothing about the complex pathway for a release or the probability of actual release of these materials to the biosphere. This is discussed in Section 5.5.

(a) Based on 0.2% uranium ore and 3% ^{235}U fresh fuel.

TABLE 7.3.17. Hazard Index of Repository Waste Inventory Relative to 0.2% Uranium Ore.[a]

Fuel Cycle	Case	Reprocessing Date	Year 2070	500 Years	1000 Years	5000 Years	10,000 Years	50,000 Years	100,000 Years	500,000 Years	1,000,000 Years
Once-Through	1	NA[b]	2.29×10^{2}	5.39	3.14	1.11	8.51×10^{-1}	4.16×10^{-1}	4.30×10^{-1}	3.74×10^{-1}	2.33×10^{-1}
	2	NA	4.38×10^{2}	7.09	3.99	1.26	9.65×10^{-1}	4.89×10^{-1}	5.23×10^{-1}	4.38×10^{-1}	2.52×10^{-1}
	3	NA	6.14×10^{2}	7.29	4.08	1.27	9.73×10^{-1}	4.93×10^{-1}	5.27×10^{-1}	4.41×10^{-1}	2.53×10^{-1}
	4	NA	7.08×10^{2}	7.33	4.09	1.27	9.72×10^{-1}	4.88×10^{-1}	5.20×10^{-1}	4.37×10^{-1}	2.51×10^{-1}
	5	NA	7.73×10^{2}	7.43	4.14	1.28	9.77×10^{-1}	4.89×10^{-1}	5.20×10^{-1}	4.37×10^{-1}	2.52×10^{-1}
Reprocessing	3	1990	5.32×10^{2}	3.26	1.63	3.07×10^{-1}	2.24×10^{-1}	8.88×10^{-2}	7.91×10^{-2}	6.58×10^{-2}	3.15×10^{-2}
		2010	5.76×10^{2}	4.16	1.90	9.56×10^{-2}	7.21×10^{-2}	3.14×10^{-2}	2.78×10^{-2}	2.57×10^{-2}	2.08×10^{-2}
	4	2000	6.30×10^{2}	3.12	1.48	1.58×10^{-1}	1.16×10^{-1}	4.22×10^{-2}	3.66×10^{-2}	3.08×10^{-2}	2.21×10^{2}
	5	2000	6.84×10^{2}	3.22	1.54	1.79×10^{-1}	1.32×10^{-1}	4.54×10^{-2}	3.88×10^{-2}	3.16×10^{-2}	2.21×10^{-2}

(a) Beyond 2070, time intervals are measured from 1980.
(b) NA = not applicable.

TABLE 7.3.18 Principal Contributors to the Hazard Index[a]

Fuel Cycle	Case	Reprocessing Date	Year 2070		1000 Years		10,000 Years		100,000 Years		1,000,000 Years	
Once-Through	3	NA[b]	^{90}Sr	95%	^{241}Am	60%	^{239}Pu	52%	^{226}Ra	65%	^{226}Ra	68%
			^{137}Cs	2%	^{240}Pu	23%	^{240}Pu	38%	^{210}Pb	22%	^{210}Pb	23%
					^{239}Pu	14%	^{226}Ra	4%	^{239}Pu	8%	^{229}Th	4%
				97%		97%		94%		95%		95%
Reprocessing	3	1990	^{90}Sr	96%	^{241}Am	75%	^{243}Am	34%	^{226}Ra	53%	^{226}Ra	30%
			^{137}Cs	2%	^{243}Am	11%	^{240}Pu	28%	^{210}Pb	18%	^{229}Th	27%
					^{240}Pu	10%	^{239}Pu	22%	^{129}I	7%	^{129}I	20%
							^{129}I	3%	^{239}Pu	6%	^{210}Pb	10%
							^{226}Ra	3%	^{237}Np	4%	^{237}Np	9%
							^{126}Sn	3%	^{229}Th	4%		
				98%		96%		93%		92%		96%
		2010	^{90}Sr	96%	^{241}Am	94%	^{243}Am	40%	^{129}I	23%	^{229}Th	41%
			^{137}Cs	2%	^{243}Am	3%	^{239}Pu	19%	^{226}Ra	22%	^{129}I	30%
							^{240}Pu	12%	^{237}Np	13%	^{237}Np	13%
							^{129}I	9%	^{229}Th	10%	^{226}Ra	9%
							^{126}Sn	7%	^{126}Sn	10%		
							^{237}Np	5%	^{210}Pb	7%		
									^{239}Pu	7%		
				98%		97%		92%		92%		93%
	4	2000	^{90}Sr	96%	^{241}Am	86%	^{243}Am	43%	^{226}Ra	31%	^{229}Th	38%
			^{137}Cs	2%	^{243}Am	8%	^{239}Pu	19%	^{129}I	18%	^{129}I	28%
							^{240}Pu	16%	^{210}Pb	11%	^{226}Ra	13%
							^{129}I	6%	^{237}Np	10%	^{237}Np	12%
							^{126}Sn	5%	^{229}Th	9%		
							^{237}Np	3%	^{239}Pu	8%		
									^{126}Sn	8%		
				98%		94%		92%		95%		91%
	5	2000	^{90}Sr	96%	^{241}Am	85%	^{243}Am	44%	^{226}Ra	33%	^{229}Th	36%
			^{137}Cs	2%	^{243}Am	9%	^{239}Pu	19%	^{129}I	17%	^{129}I	29%
							^{240}Pu	17%	^{210}Pb	11%	^{226}Ra	14%
							^{129}I	5%	^{239}Pu	9%	^{237}Np	12%
							^{126}Sn	4%	^{237}Np	9%		
							^{237}Np	3%	^{229}Th	8%		
									^{126}Sn	8%		
				98%		94%		92%		95%		91%

(a) Contribution of daughter nuclides is included.
(b) NA = not applicable.

7.4 SYSTEM RADIOLOGICAL IMPACTS

Both the regional and worldwide 70-year whole-body dose accumulations from normal operations for the proposed program, the alternative program, and the no-action alternative are compared for the once-through cycle in Table 7.4.1. Somewhat higher dose accumulations are indicated for the alternative program than for the proposed program. However, the differences are not large enough to be significant. The dose accumulation for the no-action alternative is somewhat less than for the other alternatives, but considering the time period involved, the differences are not significant. (There is a limit to how long spent fuel can be safely stored in water basins without further treatment. The assumption here is that this limit is not reached within the time frame of this analysis.) As would be expected, the dose increases with increasing size of the nuclear systems served.

TABLE 7.4.1. Comparison of 70-Year Whole-Body Dose Accumulations from Normal Operations for the Program Alternatives Using the Once-Through Cycle, man-rem

Case	Nuclear Power Growth Assumption	Proposed Program (Geologic Disposal Starting 1990 - 2010) Regional	Worldwide	Alternative Program (Disposal Starting 2010 - 2030) Regional	Worldwide	No-Action Alternative Regional	Worldwide
1	Present Inventory Only	36	48	36	48	0.2	4
2	Present Capacity Normal Life	200 to 250	290 to 370	250 to 260	370 to 380	90	160
3	250 GWe System by Year 2000 and Normal Life	940 to 1200	1400 to 1800	1200 to 1300	1800 to 1900	480	800
4	250 GWe System by Year 2000 and Steady State	1400	2100	1800	2600	NA(a)	NA
5	500 GWe system by Year 2040	1900	2800	2400	3400	NA	NA
	Dose Accumulation from Natural Radiation Sources	1×10^7	4.5×10^{10}	1×10^7	4.5×10^{10}	1×10^7	4.5×10^{10}

(a) NA = not applicable.

The regional and worldwide 70-year whole-body dose accumulations from normal operations for the proposed and alternative programs are compared for the case of reprocessing in Table 7.4.2. (The no-action alternative is not a consideration here because we assume that reprocessing would not be undertaken in that alternative.) The doses are much larger here than in the once-through cycle. However, considering the time period over which the dose is accumulated and comparing it to the dose to the regional and worldwide population that results from naturally occurring sources during the same period, 1×10^7 man-rem and 4.5×10^{10} man-rem, respectively, the dose is only a small fraction of the naturally occurring dose even in the highest nuclear growth case (Case 5); i.e., 0.5% of the regional dose

TABLE 7.4.2 Comparison of 70-Year Whole-Body Dose Accumulations from Normal Operations for the Program Alternatives Using the Reprocessing Cycle,[a] man-rem

Case	Nuclear Power Growth Assumption	Proposed Program (Geologic Disposal) Starting 1990 - 2010) Regional	Worldwide	Alternative Program (Disposal Starting 2010 - 2030) Regional	Worldwide	No-Action Alternative Regional	Worldwide
1	Present Inventory Only	NA[b]	NA	NA	NA	NA	NA
2	Present Capacity and Normal Life	NA	NA	NA	NA	NA	NA
3	250 GWe System by Year 2000 and Normal Life	13,000 to 33,000	580,000 to 970,000	13,000 to 33,000	580,000 to 970,000	NA	NA
4	250 GWe System by Year 200 and Steady State	33,000	1,000,000	33,000	1,000,000	NA	NA
5	500 GWe System by Year 2040	46,000	1,500,000	46,000	1,500,000	NA	NA
	Dose Accumulation from Natural Radiation Sources	1×10^7	4.5×10^{10}	1×10^7	4.5×10^{10}	1×10^7	4.5×10^{10}

(a) Assumed reprocessing startup dates range from 1990 to 2000.
(b) NA = not applicable.

and 0.003% of the worldwide dose. The doses from either the proposed program or the alternative program are identical. This is because the dose is accumulated primarily (about 95%) from the waste treatment operations and the same quantities of waste are treated in all cases--the only difference is that they occur at different times.

In this Statement, 100 to 800 health effects are postulated to occur in the exposed population per million man-rem. A health effect is either a fatal cancer or a genetic disorder. Based on this criterion, the program alternatives are compared on the basis of health effects in Table 7.4.3 for the once-through cycle and 7.4.4 for the reprocessing cycle. For the once-through cycle, even with the high nuclear growth assumption, the number of health effects range only from 0 to 2 on the regional basis and 0 to 3 on the worldwide basis. In the reprocessing case, the number of health effects are larger. For the high nuclear growth assumption, they range from 5 to 37 health effects on a regional basis and from 140 to 1100 on a worldwide basis. The health effects calculated to occur over the same period from naturally occurring radioactive sources range from 1000 to 8000 health effects to the regional population and 4×10^6 to 4×10^7 health effects to the worldwide population. Even though 140 to 1,100 may seem like a significant number of worldwide health effects, it is still only 0.003% of the calculated health effects to the worldwide population from naturally occuring sources of radiation over the same time period.

Neither the dose nor health effects comparison for normal operations provides a basis for favoring one of the program alternatives in either the once-through cycle or the reprocessing cycle. However, the potential impact of accidental releases might provide a basis

TABLE 7.4.3 Comparison of Normal Operations Health Effects for the Program Alternatives Using the Once-Through Cycle (number of deaths and/or genetic defects)

Case	Nuclear Power Growth Assumption	Proposed Program (Geologic Disposal) Starting 1990 - 2010)		Alternative Program (Disposal Starting 2010 - 2030)		No Action Alternative	
		Regional	Worldwide	Regional	Worldwide	Regional	Worldwide
1	Present Inventory Only	0	0	0	0	0	0
2	Present Capacity and Normal Life	0	0	0	0	0	0
3	250 GWe System by Year 2000 and Normal Life	0 to 1	0 to 2	0 to 1	0 to 2	0	0 to 1
4	250 GWe System by Year 2000 and Steady State	0 to 1	0 to 2	0 to 1	0 to 2	NA[(a)]	NA
5	500 GWe System by Year 2040	0 to 2	0 to 2	0 to 2	0 to 3	NA	NA

(a) NA = not applicable.

TABLE 7.4.4 Comparison of Normal Operations Health Effects for the Program Alternatives Using the Reprocessing Cycle (number of deaths and/or genetic defects)

Case	Nuclear Power Growth Assumption	Proposed Program (Geologic Disposal Starting 1990 - 2010)		Alternative Program (Disposal Starting 2010 - 2030)		No-Action Alternative	
		Regional	Worldwide	Regional	Worldwide	Regional	Worldwide
1	Present Inventory Only	NA[(a)]	NA	NA	NA	NA	NA
2	Present Capacity and Normal Life	NA	NA	NA	NA	NA	NA
3	250 GWe System by Year 2000 and Normal Life	1 to 26	6 to 750	1 to 26	6 to 750	NA	NA
4	500 GWe System by Year 2040	3 to 27	100 to 800	3 to 27	100 to 800	NA	NA
5	500 GWe System by Year 2040	5 to 37	140 to 1100	5 to 37	140 to 1100	NA	NA

(a) NA = not applicable.

for discrimination in the selection of a disposal program. For example, it can be argued that the longer period for research and development provided by the alternative program can in turn reduce the probability of failure by producing more knowledge and a greater diversity of choice in selecting a disposal method. Such an argument has merit only if the proposed program:

- failed to maintain R&D programs in place to increase the body of knowledge
- failed to maintain a broad base of investigation of alternative media, geology and locations so as to increase the available diversity
- failed to require technical conservatism to compensate for uncertainties and adequate factors of safety
- failed to provide for reversibility of current decisions through use of concepts of retrievability or other step-wise approaches to final decisions. This reversibility allows the increased knowledge which develops over time to be a factor in near-term decisions.

To the extent that the proposed program provides for use of the above mitigating factors, it is likely that this program would achieve safety and assurance of effective permanent disposal comparable to that of the alternative program. One purpose of including the above mitigating factors would be to make it likely that the significant long-term consequences would be indistinguishable relative to an alternative strategy.

Between similar program strategies, then, the issue becomes one of degree rather than sharp difference. Do the mitigating factors adequately compensate for the existence of uncertainties? Often such questions can only be resolved by consideration of extensive detail. In such a case, one must look to the near-term aspects of the strategies, rather than to their long-term aspects in order to evaluate significant difference which can be identified with confidence.

Reviews by the Interagency Review Group (IRG) and others indicate that the R&D program must continue to obtain necessary information before proceeding with any waste isolation concept. This program of R&D is discussed in Section 5.2 and equivalent sections throughout the Statement. Longer time spent on R&D does allow the reduction of uncertainty in understanding of key processes and parameters but generally only to a certain point. Judgments need to be made as to when sufficient R&D has been conducted and information is adequate to proceed with implementing any concept. A comprehensive discussion of the resolution of uncertainties concerning geologic disposal is contained in paragraph 2D of Appendix A to the IRG Subgroup I draft report (IRG 1979). Licensing criteria and formal consideration by DOE and by independent licensing authorities through a step-wise approach will be the mechanism for making the determination of whether enough R&D has been completed.

Any repository developed after a careful siting investigation that thoroughly examines the geological considerations discussed in Section 5.2, that proceeds in a stepwise fashion of development using technically conservative placement at each step, and that is vigorously scrutinized by independent licensing authorities should not represent a substantially greater long-term risk than any other concept.

7.5 SYSTEM RESOURCE COMMITMENTS

Estimates of required commitments for major resources for construction and operation of the entire waste management system were developed for each of the nuclear growth assumptions and for each repository and reprocessing startup date. The resources considered include steel, cement, diesel fuel, gasoline, propane, electricity and manpower. The estimated resource commitments for two cases used as reference cases for resource commitments comparisons are shown in Table 7.5.1. Resource commitments for other cases are summarized here in terms of ratios to the requirements for these reference cases. A detailed listing of these resource commitments for each case can be found in Appendix A.

The reference cases in Table 7.5.1 represent resource commitments using the Case 3 growth assumptions and a 1990 repository for the once-through cycle and a 1990 reprocessing date and a 1990 repository for the reprocessing cycle. Requirements considering all four geologic media are shown. Resource commitment variations for the different geologic media are relatively small. Requirements for reprocessing are somewhat higher than for the once-through cycle in the case of steel, cement, electricity, and manpower; are about the same to somewhat higher for diesel fuel and gasoline; and are substantially higher for propane. The higher propane requirement results from incineration of combustible waste. Gasoline and diesel fuel are used primarily in transportation. These fuel requirements are based on present practice and can be expected to change as fuel-use patterns change generally. The propane requirements for the reprocessing cycle represent about 0.5% of the total U.S. consumption for the period to year 2050 assuming current consumption rates hold constant. The largest diesel fuel use amounts to about 1% of total U.S. consumption over the period. Electricity consumption amounts of 0.02 to 0.05% to the total energy generated by the nuclear power system in this case.

The resource commitments for the program alternatives using the once-through cycle are compared in Table 7.5.2 in terms of ratios relative to the quantities in Table 7.5.1. These comparisons, which are shown as ranges, take into account the range of repository startup dates considered and the four different geologic media. In general, the requirements increase with the size of the nuclear system served. With the exception of the present inventory case, which changes only slightly, requirements for the alternative program compared to the proposed program tend to range up to 2 to 3 times higher for steel, cement, gasoline, propane, and manpower and modestly higher for diesel fuel and electricity. Requirements for the no-action alternative are zero in the present inventory case and are about the same as the alternative program for steel, cement, gasoline, propane, and manpower, but diesel and electricity consumption are much lower.

Relative resource commitments for the program alternatives in the reprocessing cycle are compared in Table 7.5.3. Requirements for the alternative program tend to be about the same to somewhat higher than the proposed program requirements.

TABLE 7.5.1 Resource Commitment Reference Cases[a]

Salt	Once-Through Cycle, 1990 Repository	Reprocessing Cycle, 1990 Reprocessing and 1990 Repository
Steel, MT	3.0×10^5	4.8×10^5
Cement, MT	2.8×10^5	5.5×10^5
Diesel Fuel, m^3	1.6×10^6	1.4×10^6
Gasoline, m^3	7.9×10^4	1.1×10^5
Propane, m^3	1.1×10^4	3.5×10^7
Electricity, kWh	6.1×10^9	1.8×10^{10}
Man Power, man-yr	8.9×10^4	1.4×10^5
Granite		
Steel, MT	4.9×10^5	7.2×10^5
Cement, MT	3.0×10^5	6.2×10^5
Diesel Fuel, m^3	1.4×10^6	1.4×10^6
Gasoline, m^3	8.6×10^4	1.5×10^5
Propane, m^3	1.3×10^4	3.5×10^7
Electricity, kWh	5.8×10^9	1.9×10^{10}
Man Power, man-yr	9.4×10^4	1.8×10^5
Shale		
Steel, MT	2.9×10^5	3.8×10^5
Cement, MT	2.9×10^5	6.4×10^5
Diesel Fuel, m^3	1.5×10^6	1.6×10^6
Gasoline, m^3	7.5×10^4	1.5×10^5
Propane, m^3	1.2×10^4	3.5×10^7
Electricity, kWh	5.4×10^9	1.9×10^{10}
Man Power, man-yr	8.6×10^4	1.8×10^5
Basalt		
Steel, MT	4.8×10^5	7.4×10^5
Cement, MT	2.7×10^5	6.1×10^5
Diesel Fuel, m^3	1.4×10^6	1.4×10^6
Gasoline, m^3	7.8×10^4	1.5×10^5
Propane, m^3	1.1×10^4	3.5×10^7
Electricity, kWh	5.8×10^9	1.8×10^{10}
Man Power, man-yr	9.9×10^4	2.0×10^5

(a) Case 3 growth assumption with 1990 repositories and 1990 reprocessing.

TABLE 7.5.2 Comparison of Relative Resource Commitments for the Program Alternatives Using the Once-Through Fuel Cycle[(a)]

Case	Nuclear Power Growth Assumption	Steel, MT	Cement, MT	Diesel, m^3	Gasoline, m^3	Propane, m^3	Electricity, kWh	Man-Power, man-yr
				Proposed Program				
1	Present Inventory Only	.02 to .05	.01 to .02	.03 to .04	.02 to .03	.02 to .03	.03	.03 to .04
2	Present Capacity with Normal Life	.29 to .77	.43 to 1.5	.18 to .26	.03 to .07	.28 to .69	.18 to .26	.26 to .48
3	250 GWe by Year 2000 with Normal Life	.97 to 3.3	.96 to 5.7	.88 to 1.2	.95 to 2.7	1.0 to 2.7	.89 to 1.2	.97 to 2.0
4	250 GWe by Year 2000 and Steady State to 2040	2.1 to 3.0	3.3 to 3.4	1.2 to 1.4	2.0 to 2.2	2.1 to 2.2	1.2 to 1.5	1.7 to 1.9
5	500 GWe System by Year 2040	2.5 to 3.7	3.6	1.6 to 1.9	2.3 to 2.5	2.5 to 2.6	1.6 to 2.0	2.2 to 2.4
				Alternative Program				
1	Present Inventory Only	.02 to .03	.008 to .02	.03 to .04	.02 to .03	.02 to .03	.02 to .03	.02 to .03
2	Present Capacity with Normal Life	.70 to .90	1.5 to 1.8	.20 to .26	.06 to .76	.62 to .80	.20 to .28	.42 to .55
3	250 GWe by Year 2000 with Normal Life	2.9 to 4.3	5.7 to 8.9	.94 to 1.3	2.4 to 3.7	2.5 to 3.9	1.0 to 1.3	1.9 to 2.7
4	250 GWe by Year 2000 and Steady State to 2040	4.0 to 4.7	8.6	1.4 to 1.7	3.7 to 3.8	3.8 to 4.1	1.3 to 1.6	2.7 to 3.0
5	500 GWe System by Year 2040	5.3 to 6.0	11.1	1.8 to 2.3	4.6 to 4.9	4.8 to 5.2	1.8 to 2.3	3.5 to 3.8
				No-Action Alternative				
1	Present Inventory Only	0	0	0	0	0	0	0
2	Present Capacity with Normal Life	.70	1.8	.07	.59	.64	.09	.36
3	250 GWe by Year 2000 with Normal Life	3.7	9.3	.04	3.2	3.4	.46	1.9

(a) Case 3 with a 1990 repository in salt was used as the reference for these ratios.

TABLE 7.5.3 Comparison of Relative Resource Commitments for the Program Alternatives Using the Reprocessing Cycle[a]

Case	Nuclear Power Growth Assumption	Steel, MT	Cement, MT	Diesel, m^3	Gasoline, m^3	Propane, m^3	Electricity, kWh	Man-Power, man-yr
				Proposed Program				
3	250 GWe by Year 2000 With Normal Life	.97 to 2.3	.96 to 3.5	.88 to 1.1	.57 to 2.3	.97 to 1.3	.89 to 1.0	.97 to 1.7
4	250 GWe by Year 2000 and Steady State to 2040	1.5 to 2.3	2.4 to 2.7	1.2 to 1.4	1.6 to 2.2	1.3	1.3	1.5 to 1.9
5	500 GWe System by Year 2040	1.7 to 2.9	2.5 to 2.9	1.6 to 2.0	2.1 to 2.7	1.7	1.7 to 1.8	1.9 to 2.5
				Alternative Program				
3	250 GWe by Year 2000 With Normal Life	1.2 to 2.9	1.1 to 3.6	.93 to 1.4	.57 to 3.5	.97 to 1.0	.94 to 1.0	1.1 to 1.9
4	250 GWe by Year 2000 And Steady State to 2040	2.0 to 2.9	2.7 to 2.9	1.6 to 1.9	2.8 to 3.3	1.3	1.3	1.6 to 2.0
5	500 GWe System by Year 2040	2.5 to 3.5	3.1 to 3.5	2.2 to 2.5	3.5 to 4.1	1.7	1.7 to 1.8	2.1 to 2.6

(a) Case 3 with 1990 reprocessing and a 1990 repository in salt was used as the reference for these ratios.

7.6 SYSTEM COSTS

Costs for the entire waste management system are presented in this section. The costs include all predisposal and disposal costs from reactor discharge of the spent fuel to final isolation of the waste in a disposal repository. The wastes include spent fuel in the once-through cycle and high-level and TRU wastes in the reprocessing cycle. The costs include the estimated expenditures by the Federal Government for research and development and repository multiple-site qualification.[a] It is assumed that these R&D costs will be recovered in accordance with the President's February 12, 1980 statement, "through fees paid by the utilities" for storage at government-owned storage facilities and for disposal at the final disposal repositories. Costs are presented here both in terms of total dollars and in terms of mills/kWh, so that the impact of this waste management on nuclear power costs can be put into perspective.

One of the most important cost components of the waste management systems is the Department of Energy's research and development and site qualification cost. The estimated annual R&D expenditures through 1995 for predisposal management of commercial wastes are tabulated in Appendix Table A.9.5. The estimated annual expenditures for disposal R&D and repository site qualification work are tabulated in Appendix Table A.9.6. Separate schedules are shown for each repository startup date considered in this analysis. The total estimated R&D and multiple site qualification costs are summarized in Table 7.6.1. These costs also include cumulative expenditures through 1980.

TABLE 7.6.1 Total Estimated Research and Development and Multiple Site Qualification Costs, $ millions

Case	Date of First Repository	Total Predisposal R&D	Total Disposal R&D and Site Verification	Total
1 & 2	1990	500	2,400	2,900
	2010	800	3,200	4,000
	2030	900	8,000	8,900
3, 4 & 5	1990	600	3,000	3,600
	2000	800	3,200	4,000
	2010	900	3,700	4,600
	2020	1,000	6,600	7,600
	2030	1,000	8,500	9,500

The R&D and multiple site qualification costs for the year 2000 repository represent an estimate for DOE's present program plan and are consistent with the program description and schedule of activities outlined in DOE's Confidence Rulemaking Statement (DOE/NE-0007

(a) "When four or five sites have been evaluated and found potentially suitable, one or more will be selected for further development as a licensed full-scale repository." President Carter, Feb. 12, 1980."

1980). (This schedule actually leads to a first repository in 1997, so some of the expenditures occur a little earlier than would be the case for a year 2000 startup.) For the 1990 repository opening, costs for activities that could not be completed by that time are deleted. Second and third repositories in 1995 and 2000 are assumed. For the 2010 repository opening, it has been assumed that the delay is caused in half by political, regulatory or other reasons at no cost and in half by technical problems with siting, licensing or other factors. Second and third repositories in 2015 and 2020 are assumed. For the 2020 and 2030 repository openings (dates within the alternative program envelope), it was assumed that expenditures continue at the 1981 level ($190 million/yr) with the program restructured to give equal emphasis to two or three disposal technologies. At the year 2000 and 2010, respectively, a preferred technology is selected and the expenditure rate is reduced by one-third. After the first repository opening (2020 and 2030, respectively), the expenditure rate is halved and continues for another 10 years when R&D is assumed to be completed.

For Cases 1 and 2, where only one repository is required, the R&D and multiple site qualification costs are reduced and phased out earlier. For the "no-action" alternative cases only the costs of R&D expended through 1980 plus the spent fuel storage R&D costs (Table A.9.5) are included, for a total of $614 million.

The total waste management costs in billions of dollars are compared for the program alternatives when using the once-through cycle in Table 7.6.2 and in Table 7.6.3 when using the reprocessing cycle. The range of costs takes into account the variation of costs with disposal and reprocessing dates and the variation in costs with the four disposal media that were considered and include the estimated R&D multiple site qualification costs. The costs increase as one would expect with the higher nuclear growth assumptions. However, they are disproportionally high for the very low growth assumptions because of the fixed costs for facilities and research and development costs. For the three cases where the no-action alternative was evaluated, the costs are similar to the low end to mid-range of the range for the proposed program. With the once-through cycle, the cost ranges are significantly higher for

TABLE 7.6.2. Comparison of Total Waste Management Costs for the Program Alternatives Using the Once-Through Cycle, $ Billions

Case	Nuclear Power Growth Assumption	Proposed Program (Geologic Disposal Starting 1990 - 2010)	Alternative Program (Disposal Starting 2010 - 2030)	No-Action Alternative
1	Present Inventory Only	5.1 to 7.6	7.4 to 14	6.4
2	Present Capacity and Normal Life	11 to 18	16 to 24	12
3	250 GWe System by Year 2000 and Normal Life	39 to 68	60 to 82	49
4	250 GWe System by Year 2000 and Steady State	61 to 72	87 to 98	NA(a)
5	500 GWe System by Year 2040	78 to 93	116 to 131	NA

(a) NA = not applicable.

TABLE 7.6.3. Comparison of Total Waste Management Costs for the Program Alternatives Using the Reprocessing Cycle,[a] $ Billions

Case	Nuclear Power Growth Assumption	Proposed Program (Geologic Disposal Starting 1990 - 2010)	Alternative Program (Disposal Starting 2010 - 2030)	No-Action Alternative
1	Present Inventory Only	NA[b]	NA	NA
2	Present Capacity and Normal Life	NA	NA	NA
3	250 GWe System by Year 2000 and Normal Life	59 to 90	58 to 90	NA
4	250 GWe System by Year 2000 and Steady State	87 to 108	89 to 104	NA
5	500 GWe System by Year 2040	114 to 146	116 to 137	NA

(a) Assumed reprocessing startup dates range from 1990 to 2010.
(b) NA = not applicable.

the alternative program than for the proposed or no-action alternatives. With the reprocessing cycle, the cost ranges are about the same for both the proposed and alternative programs.

Costs for the program alternative are compared on the basis of levelized unit costs in terms of mills/kWh at a 0% discount rate in Table 7.6.4 for the once-through cycle and Table 7.6.5 for the reprocessing cycle. On this basis, unit cost ranges for the present inventory case (Case 1) are much higher than the other cases because of the small quantity of kilowatt-hours generated in this case relative to the fixed costs. With the present capacity case (Case 2), the costs drop to about 1/3 of the Case 1 costs. For the once-through cycle, the alternative program unit costs range higher than the proposed program and the no-action alternative costs lie at the low end to mid-range of the proposed program cost range. Costs are higher for the proposed program using the reprocessing cycle than are the costs of the oncethrough cycle, but the cost range for the alternative program is almost identical to the proposed program range.

When a discount rate larger than zero is used to calculate levelized costs, the differences between the proposed program and the alternative program and differences between once-through and reprocessing cycles become less pronounced. This is shown in Tables 7.6.6 and 7.6.7, which compare the costs for the once-through cycle and the reprocessing cycle on the basis of a 7% discount rate and in Tables 7.6.8 and 7.6.9, which compare the same cost ranges on the basis of a 10% discount rate.

At a 7% discount rate, cost differences between the proposed program and the alternative program are not significant for either the once-through cycle or the reprocessing cycle. Costs for the reprocessing cycle range mostly about 10% higher to as much as 30% higher than for the once-through cycle.

At a 10% discount rate, as with a 7% rate, the cost differences between the proposed program and the alternative program are not significant. The costs for the reprocessing cycle range from slightly higher to as much as 15% higher than for the once-through cycle.

TABLE 7.6.4. Comparison of Levelized Waste-Management Unit Costs for the Program Alternatives Using the Once-Through Cycle and a 0% Discount Rate, mills/kWh[a]

Case	Nuclear Power Growth Assumption	Proposed Program (Geologic Disposal Starting 1990 - 2010)	Alternative Program (Disposal Starting 2010 - 2030)	No-Action Alternative
1	Present Inventory Only	2.9 to 4.3	4.2 to 7.7	3.6
2	Present Capacity and Normal Life	1.0 to 1.6	1.5 to 2.2	1.1
3	250 GWe System by year 2000 and Normal Life	0.7 to 1.2	1.1 to 1.5	0.9
4	250 GWe System by Year 2000 and Steady State	0.8 to 1.0	1.1 to 1.3	NA[b]
5	500 GWe System by Year 2040	0.7 to 0.9	1.1 to 1.2	NA

(a) To convert mills/kWh to $/kg HM multiply by 233.
(b) NA = not applicable.

TABLE 7.6.5. Comparison of Levelized Waste-Management Unit Costs for the Program Alternatives Using the Reprocessing Cycle and a 0% Discount Rate, mills/kWh

Case	Nuclear Power Growth Assumption	Proposed Program (Geologic Disposal Starting 1990 - 2010)	Alternative Program (Disposal Starting 2010 - 2030)	No-Action Alternative
1	Present Inventory Only	NA[a]	NA	NA
2	Present Capacity and Normal Life	NA	NA	NA
3	250 GWe System by Year 2000 and Normal Life	1.0 to 1.6	1.0 to 1.6	NA
4	250 GWe System by Year 2000 and Steady State	1.1 to 1.4	1.2 to 1.4	NA
5	500 GWe System by Year 2040	1.1 to 1.4	1.1 to 1.3	NA

(a) NA = not applicable.

A series of tables in Appendix A (Tables A.9.3a to A.9.4c) present total unit costs for each of the four geologic media over the range of 0 to 10% discount rates. These tabulations indicate generally small variations in total unit costs with the different repository media. The largest differences show up in the reprocessing cycle with early reprocessing.

Another series of tables in Appendix A (Tables A.9.1a to A.9.2c) show a breakdown of the total unit costs between spent fuel storage and transport, spent fuel treatment, other waste treatment storage and transport, disposal, and research and development. These tables show that for the once-through cycle, the research and development and site qualification cost is the dominant cost over the entire range of discount rates in the present inventory case. For the higher nuclear growth cases (cases 3, 4 and 5), research and development costs are less than 10% of the total costs at a 0% discount rate but account for one-third to one-half the cost at a 10% discount rate. Disposal costs tend to become a smaller

TABLE 7.6.6. Comparison of Levelized Waste-Management Unit Costs for the Program Alternatives Using the Once-Through Cycle and a 7% Discount Rate, mills/kWh

Case	Nuclear Power Growth Assumption	Proposed Program (Geologic Disposal Starting 1990 - 2010)	Alternative Program (Disposal Starting 2010 - 2030)	No-Action Alternative
1	Present Inventory Only	1.6 to 1.7	1.6 to 2.0	0.78
2	Present Capacity and Normal Life	0.85 to 0.92	0.87 to 1.00	0.56
3	250 GWe system by Year 2000 and Normal Life	0.61 to 0.69	0.65 to 0.68	0.49
4	250 GWe System by Year 2000 and Steady State	0.66 to 0.71	0.67 to 0.69	NA[(a)]
5	500 GWe System by Year 2040	0.64 to 0.69	0.66 to 0.67	NA

(a) NA = not applicable.

TABLE 7.6.7. Comparison of Levelized Waste-Management Unit Costs for the Program Alternatives Using the Reprocessing Cycle and a 7% Discount Rate, mills/kWh

Case	Nuclear Power Growth Assumption	Proposed Program (Geologic Disposal Starting 1990 - 2010)	Alternative Program (Disposal Starting 2010 - 2030)	No-Action Alternative
1	Present Inventory Only	NA[(a)]	NA	NA
2	Present Capacity and Normal Life	NA	NA	NA
3	250 Gwe System by Year 2000 and Normal Life	0.68 to 0.91	0.68 to 0.72	NA
4	250 GWe system by Year 2000 and Steady State	0.73 to 0.79	0.73 to 0.74	NA
5	500 GWe System by Year 2040	0.72 to 0.79	0.71 to 0.73	NA

(a) NA = not applicable.

portion of the total as the discount rate increases because they are incurred a number of years after the power is generated and thus are discounted proportionately more. In the reprocessing cycle, the research and development costs also, as in the once-throuh cycle, increase in importance as the discount rate is increased. Waste treatment and storage costs drop off significantly as the discount rate increases because these costs are deferred relative to the time of power generation. In both cycles, although spent-fuel storage and transport costs decline as the discount rate increases, they always remain a substantial portion of the total cost because they are incurred relatively soon after discharge and thus are not as heavily discounted as some of the other costs. For example, in the reprocessing cycle, spent-fuel storage and transport costs account for 30 to 60% of the total costs at a 10% discount rate compared to 20 to 50% at a 0% discount rate.

Although the total expenditure for waste management is quite large, it does not, except for the present inventory case, add more than 2 to 10%, and most likely not more than 3%, to

TABLE 7.6.8. Comparison of Levelized Waste-Management Costs for the Program Alternatives Using the Once-Through Cycle and a 10% Discount Rate, mills/kWh

Case	Nuclear Power Growth Assumption	Proposed Program (Geologic Disposal Starting 1990 - 2010)	Alternative Program (Disposal Starting 2010 - 2030)	No-Action Alternative
1	Present Inventory Only	1.2 to 1.4	1.2 to 1.4	0.61
2	Present Capacity and Normal Life	0.77 to 0.83	0.77 to 0.85	0.50
3	250 GWe system by Year 2000 and Normal Life	0.58 to 0.65	0.58 to 0.61	0.44
4	250 GWe System by Year 2000and Steady State	0.61 to 0.63	0.60 to 0.61	NA[(a)]
5	500 GWe System by Year 2040	0.60 to 0.62	0.59 to 0.60	NA

(a) NA = not available.

TABLE 7.6.9. Comparison of Levelized Waste-Management Costs for the Program Alternatives Using the Reprocessing Cycle and a 10% Discount Rate, mills/kWh

Case	Nuclear Power Growth Assumption	Proposed Program (Geologic Disposal Starting 1990 - 2010)	Alternative Program (Disposal Starting 2010 - 2030)	No-Action Alternative
1	Present Inventory Only	NA[(a)]	NA	NA
2	Present Capacity and Normal Life	NA	NA	NA
3	250 Gwe System by Year 2000 and Normal Life	0.59 to 0.77	0.59 to 0.63	NA
4	250 GWe system by Year 2000 and Steady State	0.63 to 0.66	0.63 to 0.64	NA
5	500 GWe System by Year 2040	0.63 to 0.66	0.62 to 0.63	NA

(a) NA = not available.

the total cost of nuclear power generation, which is estimated in terms of 1978 dollars to range from 25 to 35 mills/kWh for a new facility. It is also of interest to note that although the estimated expenditures for R&D and repository site qualification are very large, they amount to less than 0.5 mills/kWh (except in the present inventory case when it amounts to 2 to 5 mills/kWh at a 0% discount rate) when allocated to the generated electrical energy.

7.7 SYSTEM SIMULATION CONCLUSIONS

The system simulation analysis shows that the environmental impact of high-level and TRU waste management will be only slightly affected by waste management programs and the program strategy selected by DOE. More specifically, regarding the three program alternatives considered in this statement, the following conclusions can be drawn:

1. Radiation dose accumulations for normal operation of the required facilities increase as the size of the nuclear system increase. Neither the dose accumulation nor health effects are significantly different for the program alternatives in either the once-through or reprocessing cycles. The dose accumulation with spent fuel reprocessing is 0.5% of the regional and 0.003% of the worldwide dose from natural causes over the same period.

   For the once-through cycle, assuming continued nuclear growth, the regional 70-year whole body radiation dose accumulation over the period considered here lies in the range of 1,000 to 2,000 man-rem; an additional 400 to 1,000 man-rem are estimated for the worldwide accumulation. Comparable dose accumulations for the reprocessing cycle range from 13,000 to 46,000 man-rem for a region and 570,000 to 1,400,000 man-rem worldwide.

2. Resource commitments also increase with increasing size of the nuclear system. With the once-through cycle, resource requirements for the alternative program range up to 2 to 3 times higher than for the proposed program. With the reprocessing cycle, resource requirements for the alternative program are about the same to slightly higher than for the proposed program. Resource commitment variations relative to different geologic media are relatively small. Requirements for reprocessing are somewhat higher than for the once-through cycle for steel, cement, electricity, and manpower; about the same to somewhat higher for diesel fuel and gasoline; and substantially higher for propane. For all cases, resource requirements are a small fraction of current U.S. consumption rates.

3. Waste management costs increase with increasing size of the nuclear system but unit costs are disproportionally high for the very low-growth cases. With the once-through cycle, the cost range is significantly higher for the alternative program than for the proposed program. With the reprocessing cycle, the cost ranges are about the same for both alternatives. The no-action alternative costs are similar to the low end of the cost range for the proposed program with the once-through cycle.

   Levelized unit costs in terms of mills/kWh are sensitive to the discount rate. At a 0% discount rate, the alternative program costs are significantly higher than the proposed program costs for the once-through cycle but are about the same for the reprocessing cycle. Costs for the reprocessing cycle are higher than costs for the once-through cycle. At discount rates in the range of 7 to 10%, the differences between the proposed and alternative programs and between the once-through and reprocessing cycles become insignificant.

Unit costs for the present inventory and present capacity cases are substantially higher than for the higher nuclear growth cases because of the small amount of electricity generated relative to the fixed costs.

Assuming a 7% discount rate and continued growth of the nuclear industry, total high-level and TRU waste management costs lie in the range of 0.6 to 1.0 mill/kWh.

4. Interim storage requirements for spent fuel are substantially greater for the alternative program than for the proposed program with the once-through cycle. With the reprocessing cycle, spent fuel storage requirements are controlled by reprocessing capacity and are not sensitive to the waste management program alternatives. Storage requirements for reprocessing waste, however, become substantial with the alternative program.

   Spent fuel storage requirements are maximized with the no-action alternative.

5. Transportation requirements are higher for the alternative program compared to the proposed program with both the once-through and the reprocessing fuel cycles.

   Transportation requirements are minimized with the no-action alternative.

6. Age of the waste. A potentially beneficial aspect of the alternative program is the aging of the waste, which results in reduced radioactivity and heat generation rates which can be used to reduce repository space requirements or to further reduce the temperatures in the repository.

7. Geologic repository requirements are sensitive to the geologic medium selected, the nuclear growth rate, and the fuel cycle employed. For the highest growth assumption considered here, these requirements for operations through the year 2040 range from two to seven 800-hectare repositories for the once-through cycle and from four to nine 800-hectare repositories for the reprocessing cycle.

8. The radioactivity inventory in disposal repositories is proportional to the nuclear energy generated. The ultimate accumulation is not sensitive to the time when disposal commences but is affected by the amount of plutonium recycled and thus to the time when recycle is started.

   The inventory of fission and activation products is closely similar for both the once-through and reprocessing cycles. However, the actinide radioactivity inventory is larger for the once-through cycle than for the reprocessing cycle because all of the plutonium remains with the spent fuel. The difference in actinide activity between the two cycles is not, however, proportional to the amount of plutonium in the waste. This is because recycle of plutonium produces more of the higher actinides (e.g., americium and curium isotopes), which are discarded in the wastes. Thus, rather than a factor of 100, which could be expected on the basis of the amount of plutonium discarded, the actinide activity in the spent fuel waste is on the order of only 2 to 10 times larger than the reprocessing cycle wastes.

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CHAPTER 8

GLOSSARY OF KEY TERMS AND ACRONYMS

Abiotic: characterized by the absence of life.

Abyssal Hill: relatively small topographic feature of the deep ocean floor ranging to 600 to 900 m high and a few kilometers wide.

Actinides: Radioactive elements with atomic number larger than 88.

Activation: The process of making a material radioactive by bombardment with neutrons, protons, or other nuclear particles.

Activity: A measure of the rate at which radioactive material is emitting radiation; usually given in terms of the number of nuclear disintegrations occurring in a given quantity of material over a unit of time. The special unit of activity is the curie (Ci).

AFR: Away-from-reactor (spent fuel storage concept).

Aging: Usually refers to time to permit decay of short-lived radionuclides.

ALAP: As low as practicable, now generally replaced with ALARA (as low as reasonably achievable).

ALARA: As low as reasonably achievable. ALARA refers to limiting release and exposure and is used by the NRC (10 CFR 50.34) in the context of ". . . as low as reasonably achievable taking into account the state of technology, and the economics of improvements in relation to benefits to the public health and safety and other societal and socioeconomic considerations. . ."

Allowance Item: A number, arrived at by judgement, that represents material or equipment cost that cannot be developed otherwise because of the absence of design detail.

Alluvial Fan: A sloping, fan-shaped mass of loose rock material deposited by a stream at the place where it emerges from an upland onto a broad valley or a plain.

Alluvium: All detrital material deposited permanently or in transit by streams.

Alpha Particle: A positively charged particle emitted by certain radioactive material. It is made up of two neutrons and two protons; hence it is identical with the nucleus of a helium atom.

Amphibole: A group of dark, rock-forming, ferromagnesian silicate minerals which are closely related in crystal form and composition and which have abundant and wide distribution in igneous and metamorphic rocks.

Andesitic: A volcanic rock composed primarily of the plagioclase feldspar andesine and one or more mafic constituents.

Anion: An ion that is negatively charged.

Anticline: A fold, the core of which contains stratigraphically older rocks, which in simplest form is elongate and convex upward with the two limbs dipping away from each other.

APS: Atmospheric protection system.

Aquifer: A water-bearing layer of permeable rock or soil that will yield water in usable quantities to wells.

Aquitard: A natural rock or soil of low permeability which is stratigraphically adjacent to one or more aquifers and through which water movement is markedly retarded or impeded.

Argillaceous: Containing or pertaining to clay.

Artesian: When pertaining to an aquifer, it is one that is confined so that its hydraulic head rises above the top of the aquifer unit; thus an artesian water body is one that is confined under hydraulic pressure.

Atom: An electrically neutral particle of matter, indivisible by chemical means.

Atomic Number: The number of protons within an atomic nucleus.

Atomic Weight: The mass of an atom relative to other atoms.

Back End of the Fuel Cycle: Includes spent fuel storage, fuel reprocessing, mixed-oxide fuel fabrication, and waste management.

Background Radiation: The radiation in man's natural and undisturbed environment. It results from cosmic rays and from the naturally radioactive elements of the earth, including those from within the human body.

Basement Rock: A complex of undifferentiated rocks that underlies the oldest identifiable rocks in the area.

Basin: A depressed area generally having no outlet for surface water.

Batholith: A shield-shaped mass of igneous-intruded rock, greater than 100 km^2 in area, extending to great depth and whose diameter increases with depth.

Bedrock: A solid rock formation usually underlying one or more other loose formations.

Benthic: Refers to the bottom of a body of water.

Bentonitic: Pertaining to rock containing bentonite, a clay formed from the decomposition of volcanic ash.

Biosphere: The part of the earth in which life can exist, including the lithosphere, hydrosphere, and atmosphere; living beings together with their environment.

Biota: The animal and plant life of a region.

Biotite: A complex silicate of aluminum, potassium, magnesium, and iron with hydroxyl that is a widely distributed and important rock-forming mineral of the mica group.

Block-Faulting: A type of vertical faulting in which the crust is divided into structural or fault blocks of different elevations and orientations.

Boiling Water Reactor (BWR): A reactor system that uses a boiling water primary cooling system. Primary cooling system steam turns turbines to generate electricity.

Borosilicate Glass: A silicate glass containing at least 5 percent boric acid and used to vitrify calcined waste.

Breccia: A course-grained clastic rock composed of large, angular, and broken rock fragments cemented together in a finer grained matrix.

Burial Grounds: Areas designated for disposal of containers of radioactive wastes and obsolete or worn-out equipment by near-surface burial.

Calcine: Material heated to a temperature below its melting point to bring about loss of moisture and oxidation.

Canister: A metal container for radioactive solid waste.

Cask: A container that provides shielding and containment during transportation of radioactive materials.

Catastrophic: A violent, sudden or unexpected event which results in failure of the predicted performance of a system or component.

Cation: An ion that is positively charged.

Cation Exchange Chromatography (CEC): A process for separating several cations using the differences in the rate they travel on an ion exchange column.

Cermet: A material made by combining a heat resistant ceramic with a metal usually made by powder metallurgy.

CH-TRU: Contact-handled TRU waste.

Clastic: Pertaining to or the state of being a rock or sediment composed principally of broken fragments derived from preexisting rocks or minerals.

Colocated: Refers to location of facilities at a common site.

Concentration Guide: The average concentration of a radionuclide in air or water to which a a worker or member of the general public may be continuously exposed without exceeding radiation dose standards.

Consolidated (material): In geology, natural materials that have been made firm, cohesive, and hard.

Contact-Handled Waste: Waste package having surface dose rate less than 0.2 R/hr. Such packages can be handled by workers without extensive shielding. Contact-handled wastes were termed low-level wastes in DOE/ET-0028 and DOE/ET-0029.

Containment: Confining the radioactive wastes within presented boundaries, e.g., within a waste package.

Contingency (cost): The amount of money added to the estimated cost of a project to cover certain areas of cost uncertainty and reduce the probability of understating the project cost estimate. With the contingency added, there is a more nearly equal probability of a cost underrun or overrun.

Cost of Money: Weighted cost of debt and equity financing. Cost of money is used synonymously with cost of capital.

Critical Mass: The mass of fissionable material of a particular shape that is just sufficient to sustain a nuclear chain reaction.

Criticality: The condition in which a nuclear reactor is just self-sustaining.

Crystalline Rock: An inexact but convenient term designating an igneous or metamorphic rock, as opposed to a sedimentary rock.

Curie (Ci): A special unit of activity where 1 Ci equals 3.7×10^{10} spontaneous nuclear disintegrations per second.

Daughter Nuclide: A nuclide formed upon disintegration of a parent radionuclide.

Decommissioning: Preparations taken for retirement from active service of nuclear facilities, accompanied by the execution of a program to reduce or stabilize radioactive contamination. The objective of decommissioning is to place the facility in such a condition that future risk to public safety from the facility is within acceptable bounds.

Decontamination: The selective removal of radioactive material from a surface or from within another material.

Decontamination Factor (DF): The ratio of the original contamination level to the contamination level after decontamination.

Deep Continental Geologic Formations: Geologic media beneath the continents and isolated from the land surface by several hundred to thousands of meters of overlying rock material.

Depositional Environment (sedimentary environment): A geographically restricted environment where sediment accumulates under similar physical, chemical, and biological conditions.

Devitrification: The process by which glassy substances lose their vitreous nature and become crystalline.

Diaprisim: The piercing of overlying rocks by an upward-moving mobile core or material, such as a salt body or an igneous intrusion.

Discharge: In ground-water hydrology, water that issues naturally or is withdrawn from an aquifer.

Disposal (radioactive waste): The planned release of radioactive waste in a manner which is considered permanent so that recovery is not provided for.

Dome: A dome-shaped landform or rock mass; a large igneous intrusion whose surface is convex upward with sides sloping away at low but gradually increasing angles; an uplift or an anticlinal structure, either circular or elliptical in outline, in which the rock dips gently away in all directions, for example, a salt dome.

Dissolution: In this context it refers to the dissolving of spent fuel by nitric acid as a process step in fuel reprocessing.

Dose: Herein generally means the more rigorous term "dose-equivalent." The latter, expressed in units of rem, implies a consistent basis for estimates of consequential health risk, regardless of rate, quantity, source, or quality of the radiation exposure.

DOT: U.S. Department of Transportation.

Dry Storage: Storage of waste packages without liquid cooling.

EIA: Energy Information Administration.

EPA: U.S. Environmental Protection Agency.

Epeirogeny: The broad movements of uplift and subsidence which affect whole or large portions of continents or ocean basins.

Fault: A fracture or fracture zone along which there has been displacement of the sides relative to one another parallel to the fracture.

Fault Block: A crustal unit either completely or partly bounded by faults.

Fault System: A system of parallel or nearly parallel faults that are related to a particular deformational episode.

Feldspar: Any of a group of common rock-forming minerals that are silicates of alumina and some other base, such as potash, soda, or lime.

Fission (nuclear): The splitting of a nucleus into two or (rarely) more fragments; usually limited to heavier nuclei such as isotopes of uranium, plutonium, and thorium.

Fission Product: Any radioactive or stable nuclide produced by fission, including both primary fission fragments and their radioactive decay products.

Fissionable Material: Actinides capable of undergoing fission by interaction with neutrons of all energies.

FPF: Fuel packaging facility.

Fracture: breaks in rocks caused by intense folding or faulting or the process of breaking fluid-bearing strata by injecting a fluid under such pressure as to cause partings in the rock.

Freshwater Lens: A body of fresh water roughly shaped like a lens formed as a result of injecting freshwater into a salt water body or occurring naturally when precipitation infiltrates a saline aquifer.

Fuel (nuclear reactor): Fissionable material used as the source of power when placed in a critical arrangement in a nuclear reactor.

Fuel Cycle: Mining, refining, enrichment, and fabrication of fuel elements, use in a reactor, chemical processing to recover the fissionable material remaining in the spent fuel, reenrichment of the fuel material, refabrication of new fuel elements, and management of radioactive waste.

Fuel Element: A tube, rod, or other form into which fissionable material is fabricated for use in a reactor.

Fuel Reprocessing Plant (FRP): Plant where irradiated fuel elements are dissolved, waste materials removed, and reusable materials are segregated for reuse.

Fuel Residue Waste (FRW): Solid wastes consisting of the residue (fuel element hardware and chopped cladding material) after the bulk of fuel core material, including most of the actinides and fission products, has been dissolved in nitric acid.

Gamma Ray: Electromagnetic radiation, similar in nature to x-rays, emitted by the nuclei of some radioactive substances during radioactive decay.

GEIS: Generic Environmental Impact Statement.

Geohydrology: The study of the character, source, and mode of occurrence of underground water.

Geothermal: Pertaining to the heat of the interior of the earth.

Geothermal Gradient: The increasing temperature of the earth with depth.

GESMO: Generic Environmental Impact Statement on use of Mixed-Oxide fuel in LWRs.

Granitic: Of or pertaining to granite. Granite-like.

Granitoid: A textural term indicating grain size and mineral distribution typical of granite.

Ground Water: Water that exists or flows within the zone of saturation beneath the land surface.

Grout: A mortar fluid combined with liquid waste to provide a matrix for isolation of the waste and to seal the waste from the environment.

GWe: Gigawatts (billions of watts) of electrical generation; a rate of energy production.

Gyre: A large closed ringlike system of ocean currents which rotates in a circular motion in each of the major ocean basins.

Half-Life: a) physical--the time required for quantity of a radioactive substance to decay to one-half of its original quantity. b) biological--time required for half of an ingested or inhaled substance to be eliminated from the body by natural process. c) effective--time required for half of an ingested or inhaled radioactive substance to be eliminated from the body by the combination of radioactive decay and natural processes; mathematically equal to product of the physical and biological half-lives divided by the sum of the physical and biological half-lives.

Head End of the Fuel Cycle: Mining, milling, enrichment, and fabrication of UO_2 fuel.

HEPA: High-efficiency particulate air (filter).

High-Level Liquid Waste (HLLW): The aqueous waste resulting from operation of the first cycle solvent extraction system (or its equivalent) in a facility for reprocessing irradiated reactor fuels as well as concentrated wastes from subsequent cycles.

High-Level Waste (HLW): DOE management directives define high-level waste to include high-level liquid wastes, products from solidification of high-level liquid waste, and irradiated fuel elements if discarded without reprocessing. A proposed NRC regulation (10 CFR 60.3) defines high-level waste to include irradiated fuel, high-level liquid waste, and products from its solidification. In the GEIS there are instances, however, where discarded spent fuel and high-level waste (as wastes from the reprocessing of spent fuel) are cited separately.

Highly Enriched Uranium (HEU): Uranium containing 5% or more of added ^{235}U.

HM: Heavy metal, generally uranium and plutonium.

Hornblende: A common member of the amphibole group of minerals.

Hot Cell: A facility which allows remote viewing and manipulation of radioactive substances.

Hydraulic Gradient: The change in static head per unit of lateral distance in a given direction.

Hydrologic: Pertaining to the study of the properties, distribution and circulation of water on the surface of the land, in the soil and underlying rocks, and in the atmosphere.

Hydrostatic Pressure: The pressure exerted by the water at any given point in a body of water at rest.

ICPP: Idaho Chemical Processing Plant.

ICRP: International Commission on Radiological Protection.

ILLW: Intermediate-level liquid waste.

Immobilization: Treatment and/or emplacement of the wastes so as to impede their movement.

Interim Storage: Storage operations for which a) monitoring and human control are provided and b) subsequent action involving treatment, transportation, or final disposition is expected.

Interstices: In geology, small openings between solid particles in a rock or unconsolidated material; may be a void or pore and often contains ground water. Interstitial permeability is used to differentiate interconnected pore permeability from fracture permeability.

Ion Exchange: Replacement of ions adsorbed on a solid, such as a clay particle, or exposed at the surface of a solid by ions from solution, usually in natural water. The phenomenon is known to occur when natural water moves through clays, zeolitic rocks, and other materials of the earth's crust.

ISFS: Independent spent fuel storage.

ISFSF: Independent spent fuel storage facility.

Isolation: Segregating wastes from the accessible environment (biosphere) to the extent required to meet applicable radiological performance objectives.

Joint: A fracture or parting in a rock, along which little or no displacement of rock material has occurred.

Kaolinite: A common clay consisting mainly of hydrous aluminum silicate and closely related in chemical composition and crystal structure.

Kilowatt-hour (kWh): Use of electricity for one hour at a rate of 1000 watts.

Levelized Unit Cost: Capital and operating charges translated into an equivalent constant (or level) annual unit cost.

Light Water Reactor (LWR): May be either a BWR or PWR; uses as coolant ordinary water (H_2O) instead of heavy water (D_2O).

Lithification: The conversion of unconsolidated sediment into solid rock by processes such as compaction, cementation, and crystallization.

Lithology: The study of rocks. Also the character of a rock: its structure, color, mineral composition, grain size, and arrangement of its component parts.

Lithostatic pressure: The confining pressure at depth in the crust of the earth due to the weight of the overlying rocks.

Littoral: Belonging to, inhabiting or taking place on or near the shore of a body of water.

Low Enriched Uranium (LEU): Uranium containg less than 5% by weight but greater than 0.72% by weight ^{235}U.

M&M Shaft: Men and Materials shaft at a mined repository.

Mafic: Pertaining to or composed dominantly of magnesium rock-forming silicates.

Magmatism: The development, movement, and solidification to igneous rock, of magma, a naturally occurring mobile rock material, generated within the earth and capable of intrusion and extrusion.

Maximum Individual, Maximum-Exposed Individual: A person whose location and habits tend to maximize his radiation dose.

Megawatts (MW): Millions of watts.

Mica: A group of silicate minerals of aluminium and other bases, especially potassium, magnesium, and iron, and characterized by great perfection of cleavage in one direction, that produces thin, tough, elastic plates.

Mixed-Oxide Fuel Fabrication Plant (MOX-FFP): Plant where uranium oxide and plutonium oxide are mixed and fabricated into fuel elements for use in nuclear power plants.

MOX: Mixed oxides (of uranium and plutonium).

MTHM: Metric tons of heavy metal (usually refers to reactor fuel, in which the heavy metals are uranium and plutonium).

Mucking and/or Settling Ponds: Ponds next to drilling operations where the excavated mud or slurry is placed; the sediment that settles at the bottom of these ponds is called muck.

Multibarrier: A system using the waste form, the container (canister), the overpack, the emplacement medium, and surrounding geologic media as multiple barriers to isolate the waste from the biosphere.

NAS: National Academy of Sciences.

NASA: National Aeronautics and Space Administration.

NCRP: National Council on Radiation Protection and Measurement.

Neutron: Stable particle in a nucleus of very slightly greater mass than a proton but without nuclear change.

NO_x: Oxides of nitrogen, specifically NO and NO_2.

NRC: Nuclear Regulatory Commission.

Nucleus: The inner core of the atom, consisting primarily of neturons and protons, which make up almost the entire mass of the atom but only a minute part of its volume.

Nuclide: A species of atom characterized by its mass number, atomic number, and nuclear energy state; to be regarded as a distinct nuclide the atom must be capable of existing for a measureable lifetime in its nuclear energy state.

Olivine: An olive-green, common rock-forming ferromagnesian silicate mineral of mafic, Ultramafic, and low-silica igneous rocks.

ONWI: Office of Nuclear Waste Isolation at Battelle Memorial Intitute, Columbus, Ohio; under contract to DOE.

Operations: Broad classification of waste management activities in terms of their basic function (e.g., waste storage, treatment, transportation or disposal).

ORNL: Oak Ridge National Laboratory.

Overpack: Secondary (or additional) external containment for packaged nuclear waste.

Outcrop: A part of a body of rock that appears, bare and exposed, at the surface of the ground.

Parent Nuclide: A radionuclide that upon disintegration yields a specified nuclide, either directly or as a later member of a radioactive decay series.

Partition: To separate one (or more) element(s) from one (or more) other element(s). Examples include the separation of uranium and plutonium from each other, the separation of actinides and fission products in the waste, and the separation of one fisson product from the other fission products.

Perihelion: The point in the orbit of a celestial body that is closest to the sun.

Permeability: The quality or state of being permeable. The relative ease with which a porous medium can transmit a liquid under a hydraulic gradient.

Peridotite: A coarse-grained plutonic igneous rock composed chiefly of the mineral olivine but also containing considerable amounts of other ferromagnesian minerals.

Plagioclase: The group of common rock-forming feldspar minerals; silicates of varying mixtures of sodium and calcium.

Pluton: A body of intrusive igneous rock of any shape or size.

Pluvial: Pertaining to a period of time in which rainfall or precipitation is abundant.

PNL: Pacific Northwest Laboratory operated for DOE by Battelle Memorial Institute.

Porosity: That property of a rock or soil which enables the rock or soil to contain water in voids or interstices, usually expressed in percentage or as a decimal fraction of void volume as compared to total volume.

Pressurized Water Reactor (PWR): A reactor system that uses a pressurized water primary cooling system. Steam formed in a secondary cooling system is used to turn turbines to generate electricity.

Primary Wastes: Untreated initial wastes resulting from operation of fuel cycle facilities other than waste management facilities (wastes from operation of waste management facilities are secondary wastes).

Pyroxene: A group of dark rock-forming silicate minerals closely related in crystal form and analogous in chemical composition to the amphiboles; found chiefly in igneous rocks.

Rad: Radiation absorbed dose, the basic unit of absorbed dose of ionizing radiation. A dose of 1 rad is equivalent to the absorption of 100 ergs of radiation energy per gram of absorbing material.

Recharge: In hydrology, a source or means for replenishment of water withdrawn or discharged from an aquifer.

rem (roentgen equivalent man): A quantity used in radiation protection to express the effective dose equivalent for all forms of ionzing radiation. It is the product of the adsorbed dose in rads and factors related to relative biological effectiveness.

Remotely Handled Waste: Waste package having surface dose rate greater than 0.2 R/hr. Such packages require extensive shielding and/or remote handling to protect operating personnel. Remotely handled wastes were termed intermediate-level wastes in DOE/ET-0028 and DOE/ET-0029.

Repository (Federal): A Federally owned and operated facility for storage or disposal of specific types of waste from DOE sites and/or licensees.

Retrievability: Capability to remove waste from its place in isolation with approximately the same level of effort and radiation exposure as required to place the waste.

RH-TRU: Remotely handled TRU waste.

Risk (mathematical): Product of the consequences and the probability of the event's occurrence.

Roentgen: A unit for measuring gamma or "x-ray" radiation. The Roentgen is defined by measuring the effect of the radiation on air. It is that amount of gamma or x-rays required to produce ions carrying 1 electrostatic unit of charge in 0.001293 g of dry air under standard conditions; $1 R = 2.58 \times 10^{-4}$ coulomb/kg.

RWSF: Retrievable waste storage facility.

Scrubbers: An apparatus that chemically removes impurities from exhaust gas emissions.

Secondary Wastes: Wastes that result from applying waste treatment technologies to primary wastes.

Sedimentary Basin: A geologically depressed area that has thick sediments in the interior and thinner sediments at the edges.

Seismicity: The phenomenon of earth movements as manifested by earthquakes.

SFPF: Spent fuel packaging facility.

Shield: A continental segment of the earth's crust which has been relatively stable over a long period of time and which has exposed crystalline rocks mostly of Precambrian age; in general, representing the oldest rocks of the continent.

Shielding: A material interposed between a source of radiation and personnel for protection against the danger of radiation. Commonly used shielding materials are concrete, water and lead.

Shipping Cask: A specially designed container used for shipping radioactive materials.

SHLW: Solidified high-level waste.

Short-Lived Nuclides: Radioactive isotopes with relatively short half-lives. Usage for some isotopes varies with the concept being considered (e.g., isotopes with 5-50 year half-lives are short lived in the context of geologic disposal but long lived in the context of predisposal operations).

Slurry: A fluid mixture or suspension of insoluble material.

Solidification: Conversion of liquid radioactive waste to a dry, stable solid.

Source Terms: The quantity of radioactive material (or other pollutant) released to the environment at its point of release (source).

Spent Fuel (SF): Nuclear reactor fuel that has been used to the extent that it can no longer be used efficiently in a nuclear power plant.

Stock: An igneous intrusion less than 100 km^2 (40 mi^2) in surface exposure.

Storage: Retention of waste in some type of manmade device in a manner permitting retrieval.

Strain: Deformation resulting from applied stress; proportional to stress.

Stratum: Sedimentary bed or layer, regardless of thickness, of homogeneous or gradational lithology.

Syncline: A fold, the core of which contains stratigraphically younger rocks, and which, in simplest form, is elongate and concave upward with the two limbs dipping toward each other.

Tailings: The part of any ore that is regarded as too poor to be treated further.

Tails: In the case of uranium it refers to the depleted uranium left after enrichment operations.

TBP: Tributyl phosphate, a solvent used in the PUREX fuel reprocessing process.

Technologies: Specific methods for implementing concepts. An example is calcination of liquid high-level waste by using a spray calciner.

Tectonic: Of, pertaining to, or designating the processes causing, and the rock structures resulting from, deformation of the earth's crust.

Tectonism (diastrophism): Crustal movement produced by earth forces, such as the formation of plateaus and mountain ranges; the structural behavior of an element of the earth's crust during, or between, major cycles of sedimentation.

Theoretical Density (TD): Maximum density attainable for any given material.

Thermal Regime: The area adjacent to a heat source which is affected by that source.

Trajectory: The curve that an object describes in space in traveling from one point to another.

Transmissivity: Volume of water flowing through a 1-ft width of aquifer of given thickness under a unit gradient (1 ft vertically for each 1 ft laterally) and at the viscosity prevailing in the field. Mathematically, it is the product of permeability and aquifer thickness.

Transmutation: A nuclear process in which one nuclide is transformed into the nuclide of a different element. This can be accomplished by bombardment with neutrons or other nuclear particles.

Transportation: Movement of materials between sites. Intra-site movement is not considered. Includes alternative methods for packaging, handling, and transport of waste materials and plutonium compounds. Concepts include all conventional methods of land and water transport required by the waste management system.

Transuranic (TRU) elements: Elements with atomic number greater than 92. They include, among others, neptunium, plutonium, americium, and curium.

Transuranic Waste: Waste material measured or assumed to contain more than a specified concentration of transuranic elements. For purposes of this Statement, TRU waste is waste from locations that might cause contamination levels above 10 nanocuries of transuranic alpha activity per gram of waste.

Treatment: Operations intended to benefit safety or economy by changing the waste characteristics.

Ultramafic: Pertaining to igneous rocks composed chiefly of ferromagnesian dark minerals.

Uplift: A structurally high area in the crust, produced by movements that raise or upthrust the rocks, as in a dome or arch.

Vital Areas: The code of Federal Regulations (10 CFR 73), defines equipment items, systems, devices, and materials whose failure, destruction or release could directly endanger the public health and safety by exposure to radiation defined as "vital". Areas containing such items or materials (e.g., spent fuel or high-level waste) are defined as "vital" areas and subject to special protection measures.

Waste Immobilization: Process of converting waste to a stable, solid and relatively insoluble form.

Waste Isolation Pilot Plant (WIPP): A Defense repository proposed for a site in Southeastern New Mexico.

Waste Management: The planning, execution and surveillance of essential functions related to the control of radioactive (and nonradioactive) waste, including treatment, transportation, storage, surveillance, and isolation.

Water Table: The upper surface of the zone of water saturation in the subsurface, at which the pressure is equal to atmospheric pressure; the upper surface of an unconfined aquifer.