



Department of Energy

Germantown, MD 20874-1290

JUL 26 1996

RECEIVED BY E.P. STROUPE	
AUG 5 1996	

Mr. Elwood P. Stroupe
Lockheed Idaho
1977 Freemont Avenue
Idaho Falls, Idaho 83401-3135

Dear Mr. ^{Woody} ~~Stroupe~~:

I wish to take this opportunity to thank you for your support in preparing the *Technical Strategy for the Treatment, Packaging, and Disposal of Aluminum-Based Spent Nuclear Fuel*. This report of the Research Reactor Spent Nuclear Fuel Task Team was prepared in a diligent and timely manner and reflects positively on the professionalism you have shown the Department of Energy. The Task Team's recommendations have been reviewed by the Department and the path forward approved by the Assistant Secretary of Environmental Management. We now look forward to the implementation of this challenging program.

Enclosed are copies of the final report (Volumes I and II). Should you desire additional copies or have inquiries as to how our program is progressing, please contact me or Howard Eckert at (301) 903-1450.

Sincerely,

G. Frank Cole, Director
Office of Spent Fuel Management
Office of Nuclear Material and
Facility Stabilization
Office of Environmental Management

Enclosure



Summary

FINAL ENVIRONMENTAL IMPACT STATEMENT



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



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

Copies are available (while supplies last) from DOE's:

Center for Environmental Management
470 L'Enfant Plaza East, SW
Suite 7112
Washington, DC 20024

1-800-7-EM-DATA (1-800-736-3282)

Jim Linhart



Department of Energy

Washington, DC 20585

February 8, 1996

Dear Interested Party:

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

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

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

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

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

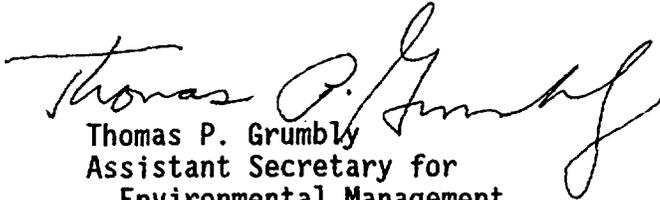


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Center for Environmental Management Information at 1-800-736-3282 (toll-free). The entire document will be placed in the public reading rooms and information locations listed in the Summary.

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

Sincerely,


Thomas P. Grumbly
Assistant Secretary for
Environmental Management

Enclosure

Summary

FINAL ENVIRONMENTAL IMPACT STATEMENT

on a

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



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

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

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

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

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Summary

S.1 Introduction

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

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

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

National Environmental Policy Act

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

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

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

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

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

S.1.1 Policy Background

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

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

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

What is Spent Nuclear Fuel?

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

What is Enriched Uranium?

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

Uses and Benefits of Research Reactors

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

Advances in Nuclear Medicine



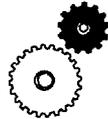
Cancer therapy, medical isotope production, clarification of the biological effects of radiation, development of improved drugs, and blood testing.

Environmental, Agricultural, and Climate Studies



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

Benefits to Industry



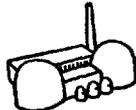
Neutron radiography allows diagnosis of defects in metals and engines, research on new and improved materials, and leak detection.

Advancement of Basic Scientific Research



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

Nonproliferation



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

Materials and Advanced Fuels Testing



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

Figure S-1 Uses and Benefits of Research Reactors

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

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

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

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

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

first two Management Alternatives. In Management Alternative 1, the Draft EIS assesses the impacts of managing the spent nuclear fuel at five DOE sites and using ten candidate ports of entry.

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

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

S.1.2 Purpose and Need for Agency Action

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

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

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

The Treaty on the Non-Proliferation of Nuclear Weapons

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

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

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

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

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

S.1.3 Decisions to be Made Based on this EIS

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

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

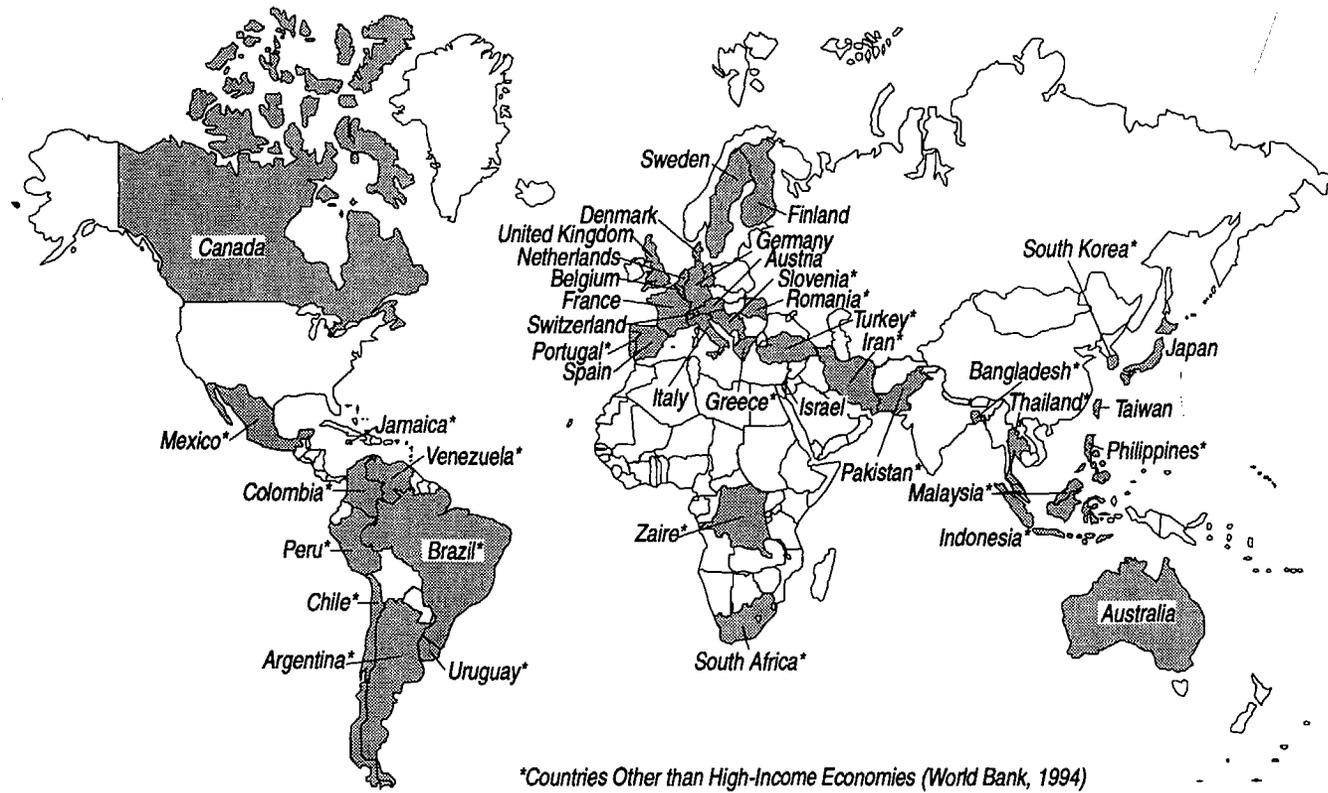


Figure S-2 Countries which Host the Research Reactors

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

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

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

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

S.2 Proposed Action and Alternatives

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

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

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

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

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

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

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

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

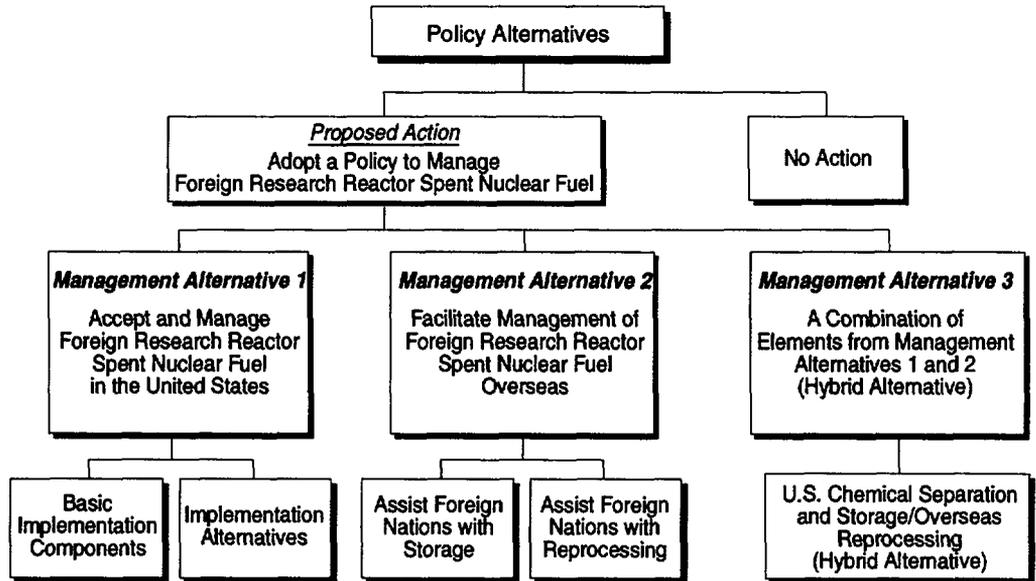


Figure S-3 Management Alternatives of the Proposed Action

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

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

The three Management Alternatives are summarized below.

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

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

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

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

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

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

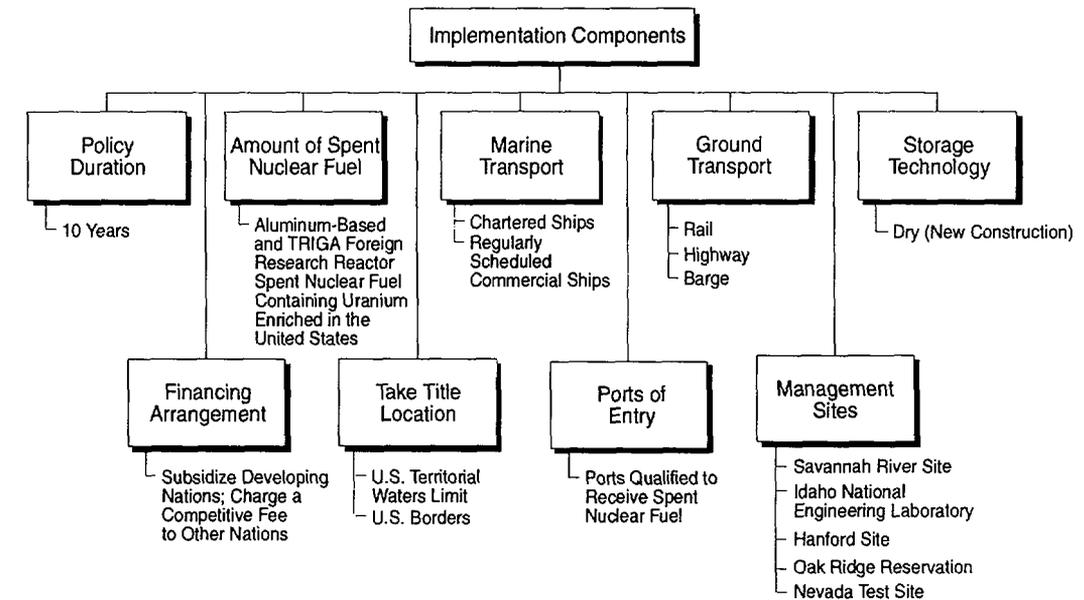


Figure S-4 Basic Implementation Components of Management Alternative 1

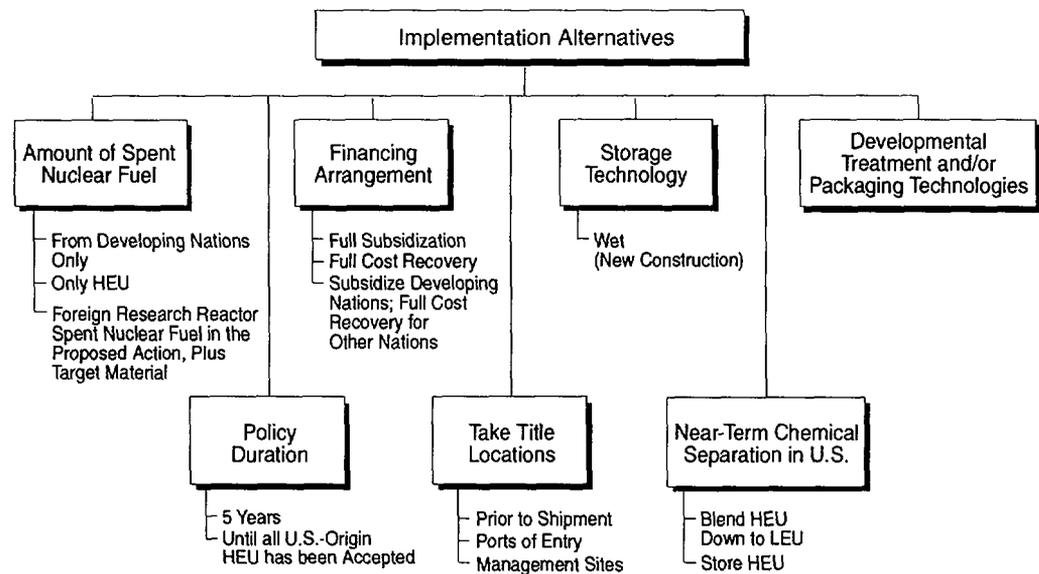


Figure S-5 Implementation Alternatives

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

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

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

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

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

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

S.2.2.1 Basic Implementation Components

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

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

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

S.2.2.1.1 Policy Duration

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

S.2.2.1.2 Financing Arrangement

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

S.2.2.1.3 Amount of Foreign Research Reactor Spent Nuclear Fuel

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

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

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

S.2.2.1.5 Marine Transport

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

As a comparison:

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

S.2.2.1.6 Port(s) of Entry

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

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

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

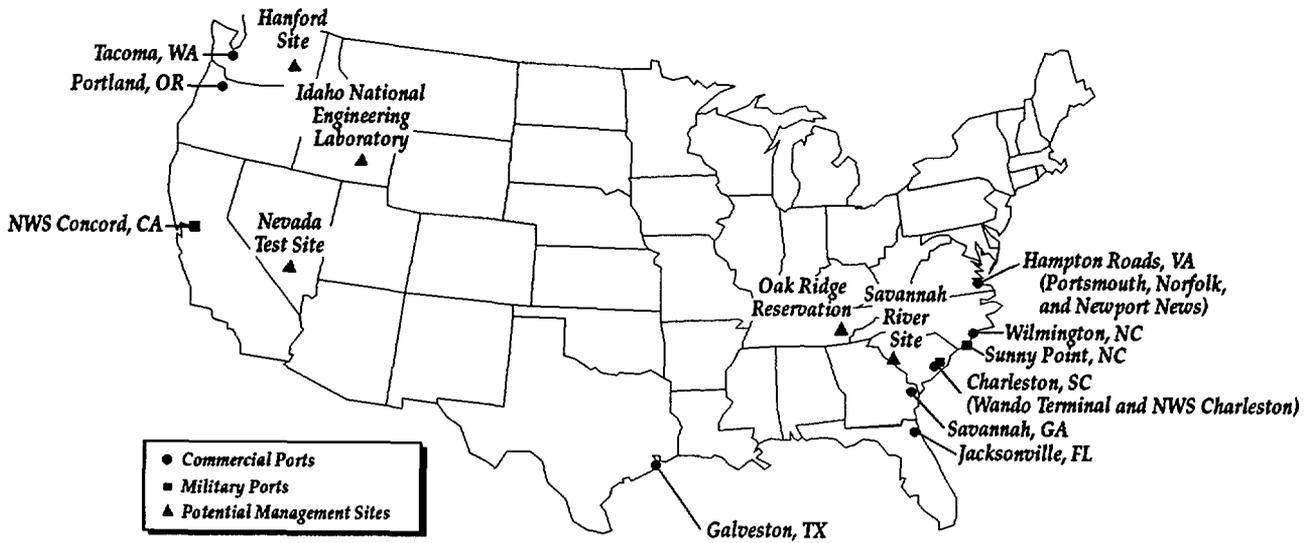


Figure S-6 Location of Potential Ports of Entry and Management Sites

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

S.2.2.1.7 Ground Transport

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

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

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

S.2.2.1.8 Foreign Research Reactor Spent Nuclear Fuel Management Sites

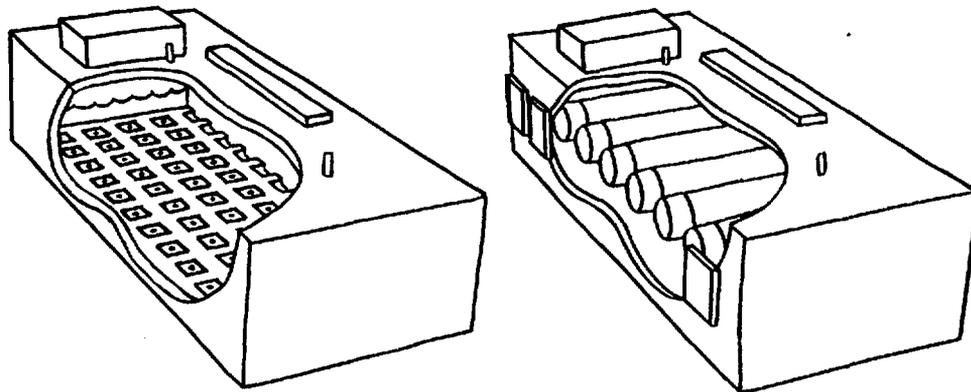
Potential sites for the receipt and management of foreign research reactor spent nuclear fuel have been specified by DOE in the Record of Decision for the Programmatic SNF&INEL Final EIS, which is concerned with the environmental impacts of management of spent nuclear fuel. In accordance with this Record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

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

S.2.2.1.9 Storage Technologies

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

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



Typical Wet Storage Facility

Typical Dry Storage Facility

Figure S-7 Typical Storage Facilities

S.2.2.1.10 Ultimate Disposition

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

S.2.2.2 Implementation Alternatives for Management Alternative 1

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

1. Alternative amount of foreign research reactor spent nuclear fuel to be accepted:
 - a. Only from countries with other-than-high-income-economies (up to 1.9 MTHM; 5,000 elements)
 - b. HEU only (up to 4.6 MTHM; 11,200 elements)
 - c. Target material in addition to spent nuclear fuel (up to 0.6 MTHM; equivalent to 620 elements)
2. Alternative policy durations:
 - a. Five-year policy (up to 13 MTHM; 18,800 elements)
 - b. Indefinite HEU/10-year LEU policy (same amount as basic implementation; different timing)
3. Alternative financing arrangements:
 - a. Subsidize all countries

- b. Charge all countries full cost of accepting and managing foreign research reactor spent nuclear fuel
 - c. Subsidize other-than-high-income economy countries; charge high-income economy countries full-cost recovery fee
4. Alternative locations for taking title:
 - a. Prior to shipment
 - b. Port(s) of entry
 - c. Management sites
 5. Wet storage technology for new construction
 6. Near-term conventional chemical separation in the United States¹
 - a. Extent of chemical separation: dedicated to foreign research reactor spent nuclear fuel only, or part of larger-scale DOE chemical separation activities
 - b. Uranium disposition: blend HEU down to LEU or process HEU to oxide for interim storage
 7. Developmental treatment and/or packaging technologies (Conduct a development program leading to a decision on whether to construct and operate a cost-effective new treatment and/or packaging facility. The objective of this technical strategy is to treat, package, and store spent nuclear fuel in a manner suitable for placement into a geologic repository.

S.2.3 Preferred Alternative

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

1. U.S. Government nuclear weapons nonproliferation policies and objectives;
2. DOE responsibilities (e.g., safe handling of hazardous materials, safety/health risks to workers, compatibility with other ongoing missions, etc.);
3. Potential environmental impacts (e.g., public safety, etc.);
4. Public comments received and concerns expressed following issuance of the Draft EIS;
5. Analysis of impacts and alternatives in the Programmatic SNF&INEL Final EIS (DOE, 1995c), as well as the Record of Decision for that EIS;

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

6. Estimated costs of alternatives for management of foreign research reactor spent nuclear fuel;
7. Public issues/concerns/perceptions (e.g., fairness/equity to affected States and populations, etc.); and
8. Uncertainties (e.g., future budget priorities and continuity of funding, technology development, repository timing and waste form acceptance criteria, regulatory change, etc.).

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

The amount of foreign research reactor spent nuclear fuel that would be accepted and managed, as specified in Section 2.2.1.3 of the EIS, could total approximately 19.2 MTHM, with a volume of approximately 110 m³ (4,100 ft³), representing approximately 22,700 individual spent nuclear fuel elements. The target material that would be accepted and managed, as specified in Section 2.2.2.1 of the EIS, contains an additional 0.6 MTHM representing the uranium content of approximately 620 additional typical foreign research reactor spent nuclear fuel elements. The following stipulations on qualifying spent nuclear fuel types would apply:

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

Preferred Alternative Elements

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

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

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

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

Marine Transport: Either chartered or commercial ships.

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

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

Ground Transport: Truck or rail.

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

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

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

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

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

- Spent nuclear fuel (HEU) would be accepted from research reactors having lifetime cores, from research reactors planning to shut down by a specific date while the policy is in effect, and from research reactors for which a suitable LEU fuel is not available.
- Spent nuclear fuel (HEU and/or LEU) would be accepted from research reactors that are already shut down.
- Unirradiated fuel (HEU and/or LEU) from eligible research reactors would be accepted as spent nuclear fuel.
- For research reactors with both HEU and LEU spent nuclear fuel available for shipment, LEU spent nuclear fuel would not be accepted until the HEU spent nuclear fuel is exhausted, unless there are extenuating circumstances (e.g., deterioration of one or more LEU elements sufficient to cause a safety problem).
- Spent nuclear fuel (HEU and/or LEU) would not be accepted from new research reactors starting operation after the date of implementation of the policy.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

1. ***Management Alternative*** - The various management alternatives considered are discussed in Sections 2.2 through 2.4 of the EIS and S.2.2, S.2.4 and S.2.5 of the Summary. The analyses in Sections 4.2 through 4.5 of the EIS and S.4.2, S.4.3, S.4.5 and S.4.6 of the Summary demonstrate that the impacts on the environment, involved workers, or the citizens of the United States from implementation of any of the management alternatives or implementation alternatives analyzed (other than beneficial impacts associated with support for United States nuclear weapons nonproliferation policy) would be small and completely within the applicable regulatory limits, and would not provide a basis for discrimination among the alternatives. As a result, the process for selection of the elements of the preferred alternative focused on programmatic considerations:
 - a. DOE and the Department of State concluded that the No Action Alternative and Management Alternative 2, Implementation Alternative 1a (Overseas Storage) would be unacceptable since these alternatives are not consistent with United States nuclear weapons nonproliferation policy objectives.
 - b. DOE and the Department of State believe that the basic implementation of Management Alternative 1 would be undesirable to the extent that it would involve indefinite storage of foreign research reactor spent nuclear fuel in a form that is not suitable for disposal. Management Alternative 1 modified to rely solely on Implementation Alternative 6 (Near Term Conventional Chemical Separation in the United States) would raise nuclear weapons nonproliferation policy questions. Management Alternative 1 modified to rely solely on Implementation Alternative 7 (Developmental Treatment and/or Packaging Technologies) could not be selected at this time because no decision has been made on which technology will be pursued.
 - c. DOE and the Department of State also believe that Management Alternative 2, Implementation Alternative 1b (Overseas Reprocessing) would be technically complex and potentially extremely expensive because it would require the United States to accept reprocessing wastes from the overseas reprocessing operations. This is due to the fact that both of the countries in which the overseas reprocessing might be accomplished require the reprocessing wastes to leave their countries, and many of the countries that would be covered by the proposed policy cannot accept the return of such reprocessing wastes. The intermediate-level radioactive wastes produced in Europe during reprocessing of research reactor spent nuclear fuel are often in a concreted waste form, unlike any high-level radioactive waste form in the United States. This concreted waste form has not been evaluated for disposal in a United States geologic repository. Accordingly, acceptance of such waste in the United

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

1a. Overseas Storage

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

1b. Overseas Reprocessing

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

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

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

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

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

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

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

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

S.2.6 No Action

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

S.2.7 Characteristics of Spent Nuclear Fuel Management

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

S.2.7.1 Characteristics of Foreign Research Reactor Spent Nuclear Fuel

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

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

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

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

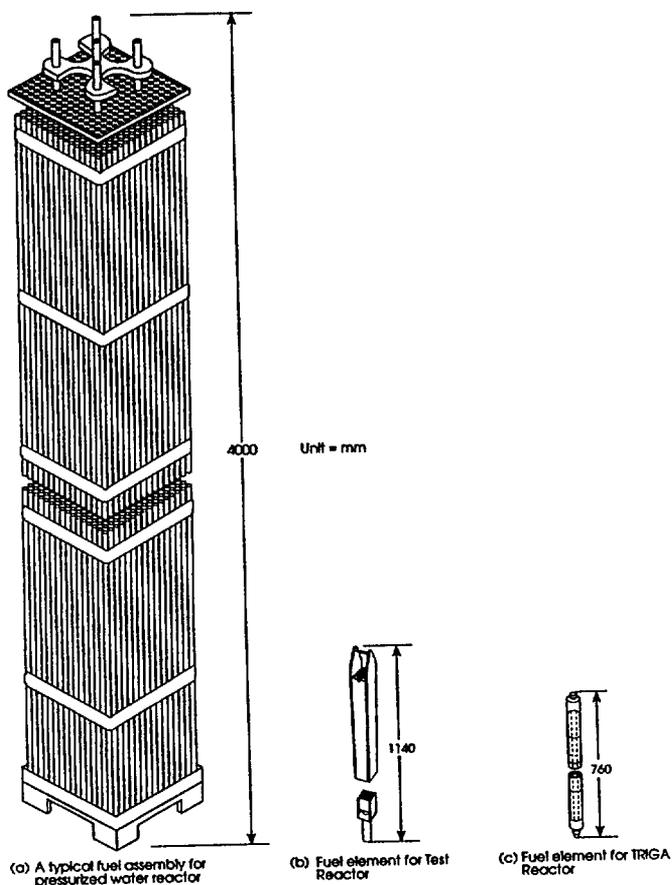


Figure S-8 Typical Spent Nuclear Fuel Elements

S.2.7.2 Transportation Casks

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

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

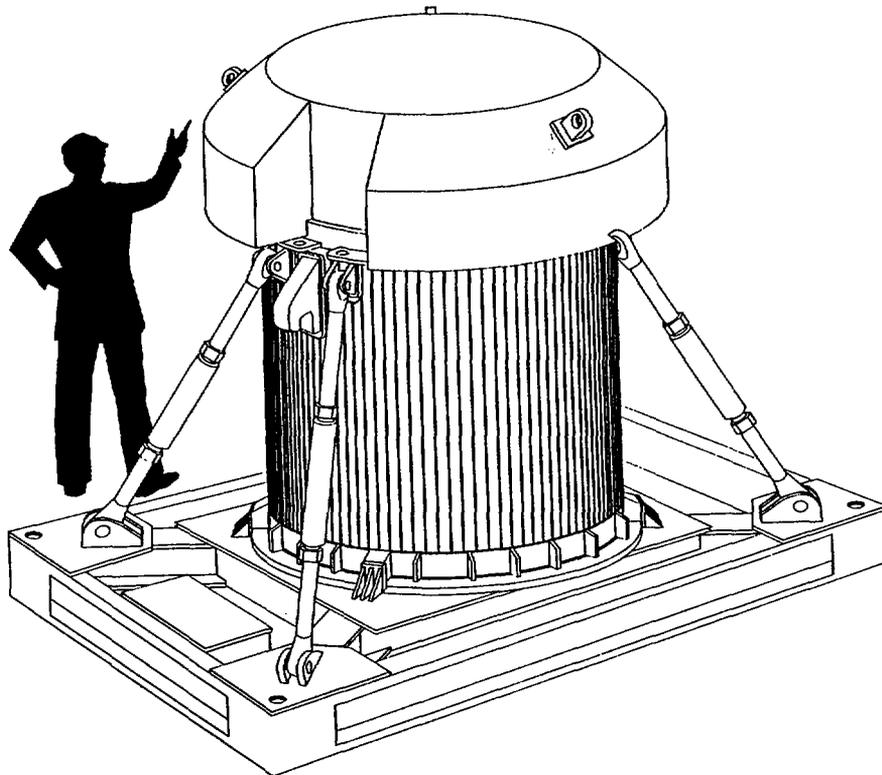


Figure S-9 Typical Spent Nuclear Fuel Transportation Cask

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

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

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

facilities could be used (Phase 2). The following discussion summarizes key points concerning facility capabilities and assumptions at each site, which drive the analysis of environmental impacts in the EIS.

S.2.7.3.1 Savannah River Site

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

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

S.2.7.3.2 Idaho National Engineering Laboratory

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

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

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

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

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

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

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

S.2.7.5 Foreign Reprocessing Facilities

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

S.2.8 Emergency Management and Response

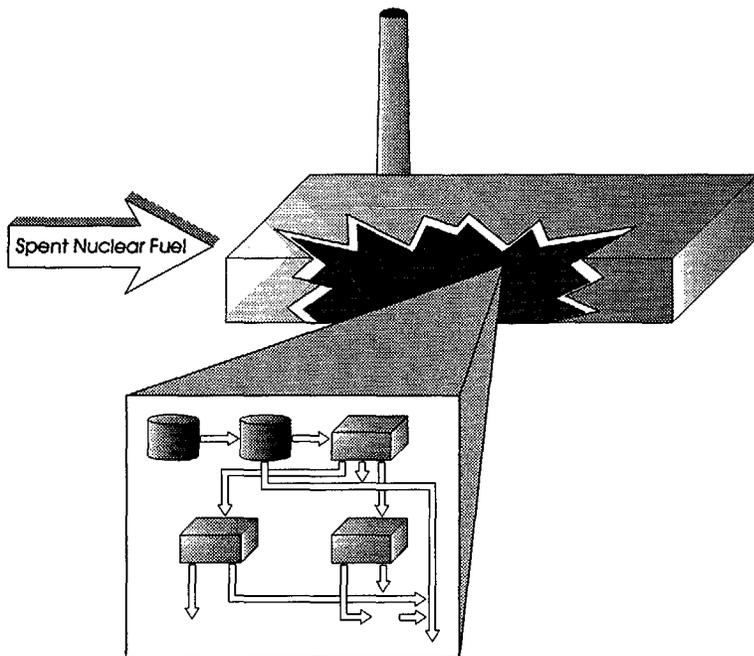
An emergency management and response infrastructure exists to support the implementation of those Management Alternatives that would be carried out in the United States, including ports of entry, ground transport routes, and management sites. In the United States, State and local governments are required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

S.2.9 Security Measures

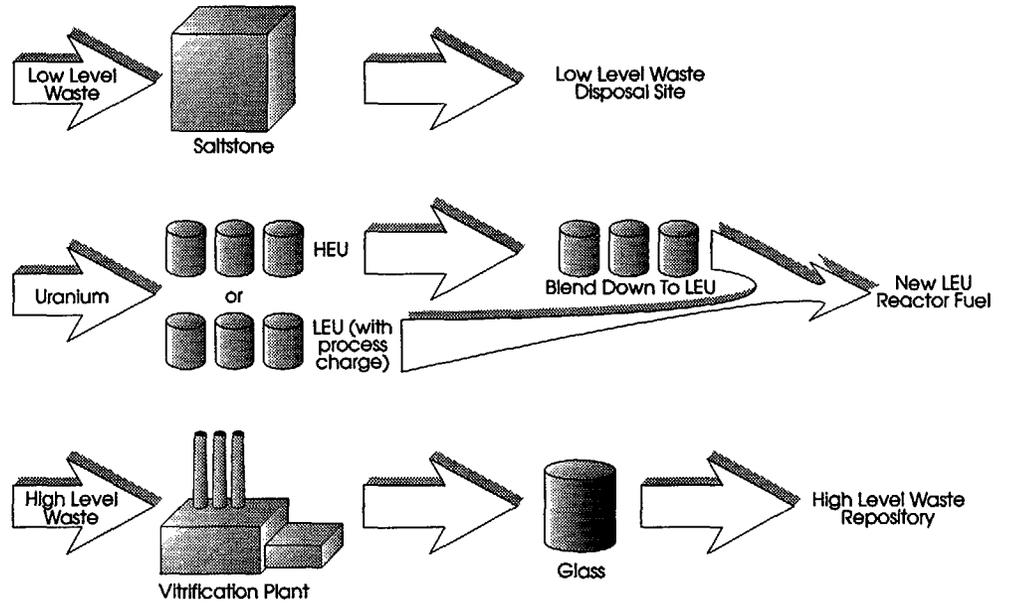
Domestic transportation of foreign research reactor spent nuclear fuel would be under the regulatory jurisdiction of the Department of Transportation and the NRC. In the event that foreign research reactor spent nuclear fuel was transported through a military port of entry, applicable requirements would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

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

Chemical Separation Process



Chemical Separation Products



S U M M A R Y

Figure S-10 Chemical Separation Overview

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

S.2.10 Additional Alternatives Considered but Dismissed from Detailed Analysis

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

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

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

S.3 Affected Environment

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

S.4 Policy Considerations and Environmental Impacts

The EIS assesses the policy considerations and potential environmental impacts resulting from each of the Management Alternatives for implementation of the proposed policy, including the preferred alternative designated by DOE and the Department of State, and the No Action Alternative. Policy considerations are addressed by characterizing the extent to which each of the alternatives supports the U.S. goal of nuclear weapons nonproliferation. This characterization takes two forms: 1) estimates of the maximum

amount of HEU that could be removed from international commerce under each alternative, and 2) the extent to which each alternative would provide incentives for foreign research reactor operators to convert their reactors to LEU fuel.

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

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

S.4.1 Overview of Environmental Impacts

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

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

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

S.4.1.1 General Radiological Health Effects

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

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

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

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

Measuring Radiation Exposure

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

S.4.2.1 Marine Transport Impacts

The shipment of foreign research reactor spent nuclear fuel would begin with the transport of the spent nuclear fuel from the onsite storage facility at the foreign research reactor to the foreign port. The spent nuclear fuel would then be shipped in transportation casks by sea (except for shipments from Canada) to a U.S. port. The potential impact of marine transport in the territorial waters of the United States was evaluated. Because

implementation of the proposed action could involve ocean transport, the EIS also considers the environmental impacts on the global commons in accordance with Executive Order 12114. Shipments of any material via ocean transport entails risks to the ship's crew members and the environment. The risks result from transportation-related accidents and, in the case of radioactive materials, from exposure to the material itself.

S.4.2.1.1 Impacts of Incident-Free Marine Transport

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

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

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

S.4.2.1.2 Impacts of Accidents During Marine Transport

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

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

S.4.2.2 Port Activities Impacts

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

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

S.4.2.2.1 Impacts of Incident-Free Port Activities

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

S.4.2.2.2 Impacts of Accidents During Port Activities

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

S.4.2.3 Ground and Barge Transport Impacts

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

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

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

S.4.2.3.1 Impacts of Incident-Free Ground Transport

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

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

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

S.4.2.3.2 Impacts of Accidents During Ground Transport

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

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

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

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

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

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

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

S.4.2.4.1 Impacts from Incident-Free Management Site Activities

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

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

The highest estimated population risk to the workers who perform foreign research reactor spent nuclear fuel handling operations would be 0.21 latent cancer fatalities resulting from the combination of the near-term receipt and storage of foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory, and the future receipt and

storage at a different site. The maximum exposure to an individual worker was not calculated due to the large uncertainties involved with such calculations. A very conservative upper bound, however, would be the regulatory limit of 5,000 mrem per year, which translates to 0.026 latent cancer fatalities for workers receiving such a dose for the 13-year period during which receipts could take place.

S.4.2.4.2 Impacts of Accidents During Management Site Activities

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

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

S.4.2.4.3 Other Potential Environmental Impacts from Management Site Activities

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

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

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

Cultural Resources. Although most of the potential management sites contain areas of archaeological, cultural, or historic interest, little or no direct impacts on cultural resources would be expected because of the location of the storage facilities. However, site surveys

would be conducted prior to construction. In the event that cultural resources were found, the State Historic Preservation Officer would be contacted. Tribal leaders would be contacted if any Native American resources were found.

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

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

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

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

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

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

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

spent nuclear fuel management activities, including foreign research reactor spent nuclear fuel storage facilities. However, all other existing system capacities would manage the estimated increases for materials, utilities, and energy.

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

S.4.2.4.4 Cumulative Impacts at the Management Sites

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

S.4.2.4.5 Impacts of Ultimate Disposition

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

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

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

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

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

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

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

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

S.4.3.1 Implementation Alternative 1: Different Amounts of Material

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

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

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

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

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

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

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

S.4.3.2 Implementation Alternative 2: Alternative Policy Durations

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

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

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

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

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

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

S.4.3.3 Implementation Alternative 3: Alternative Financing Arrangements

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

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

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

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

The environmental impacts of the proposed foreign research reactor spent nuclear fuel program are not affected by who the owner of the spent nuclear fuel is or the point at which title is transferred. The Price-Anderson Act would apply to the spent nuclear fuel shipments, once they arrive in territorial United States, regardless of who holds title to the

fuel. Thus, there would be no change in the liability protection provided to the citizens of the United States, no matter where DOE would take title. Ownership would not affect shipping arrangements and precautions, or liability protection, other than to increase DOE's potential liability if DOE were to take title before shipment.

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

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

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

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

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

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

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

The principal environmental impacts under this implementation alternative would be occupational and public health and safety impacts. The total incident-free population risk to the worker population of incurring a latent cancer fatality resulting from this implementation alternative would be 0.32 latent cancer fatalities among all marine, port, ground transport, and site workers combined. The largest contribution to this total risk

would be from onsite radiation workers. The risk to onsite radiation workers would be 0.21 latent cancer fatalities, which translates to 21 chances in 100 that one worker within the group of exposed workers would develop a latent cancer fatality.

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

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

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

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

S.4.4 Implementation of the Preferred Alternative

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

would be necessary. All the TRIGA foreign research reactor spent nuclear fuel, representing approximately 162 casks and 4,900 elements would be transported to and managed at the Idaho National Engineering Laboratory.

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

S.4.4.1 Policy Considerations

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

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

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

DOE is aware that the inclusion of chemical separation within the preferred alternative could be interpreted by some nations, organizations, and persons as a signal of endorsement of the use of chemical separation as a routine method of waste management for spent nuclear fuel or a reversal of United States policy on chemical separation. This would not be an accurate interpretation. The United States policy regarding chemical

separation was established in Presidential Decision Directive 13, and DOE and the Department of State have determined that this preferred alternative is consistent with that policy. The draft version of this EIS indicated that chemical separation is a non-preferred technology. This final preferred alternative includes provision for possible chemical separation. DOE maintains a presumption that spent nuclear fuel would not be chemically separated unless there is an imminent health and safety risk, or other programmatic conditions, that cannot be addressed during the time period when no feasible alternative to chemical separation is available. These conditions will be addressed by the independent study described in S.2.3.

S.4.4.2 Potential Environmental Impacts

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

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

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

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

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

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

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

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

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

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

If the United States does not accept any near-term foreign research reactor spent nuclear fuel shipments, provision of U.S. technical and/or financial assistance for the development of safe and secure storage capabilities would help to alleviate some of the problems posed by a lack of sufficient storage capacity. However, this subalternative presents several

drawbacks from a nuclear weapons nonproliferation policy standpoint. The accumulation overseas of ever larger amounts of spent nuclear fuel containing HEU poses a risk that such weapons-usable material might be illicitly diverted to a weapons program. Although U.S. assistance in maintaining adequate physical security for foreign research reactor spent nuclear fuel repositories may lessen the potential for diversion, the proliferation risk would still be greater than under the basic implementation of Management Alternative 1. As the foreign research reactor spent nuclear fuel ages, it would become less radioactive and thus a more attractive target for illicit diversion.

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

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

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

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

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

S.4.7 No Action Alternative

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

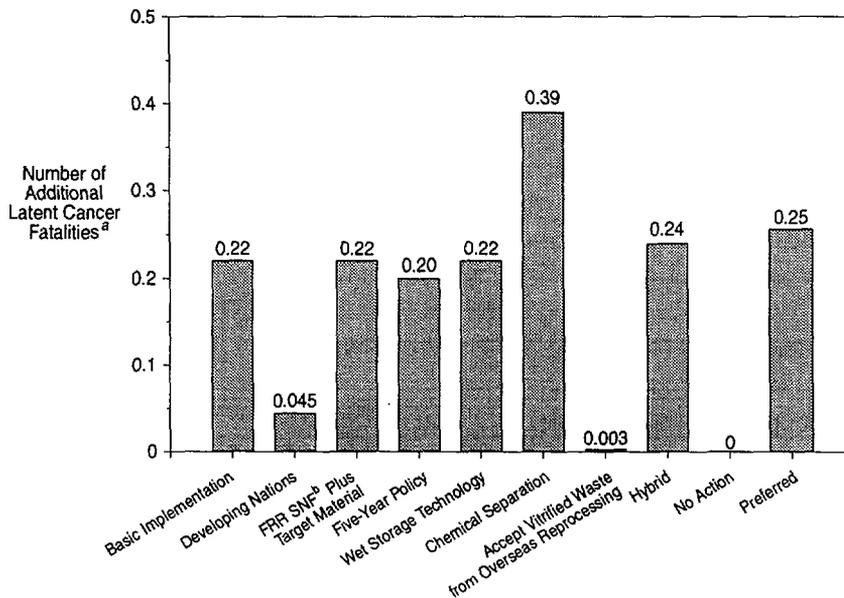
S.4.8 Comparison of the Radiological Risks

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

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

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

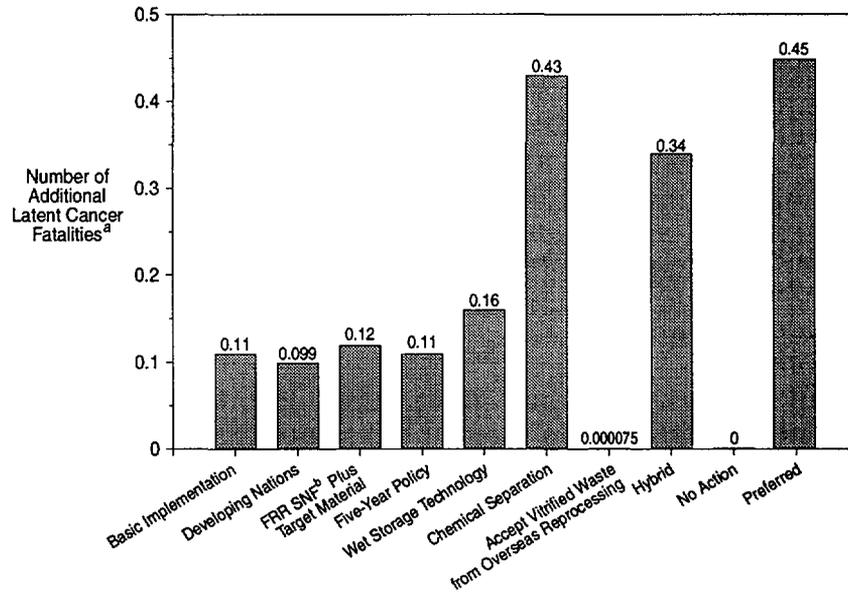
Incident-free population risks for workers are depicted in Figure S-13. The greatest incident-free radiological population risk to workers from any of the alternatives would occur in the alternative in which target material is added to the basic implementation of



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

^b Foreign Research Reactor Spent Nuclear Fuel

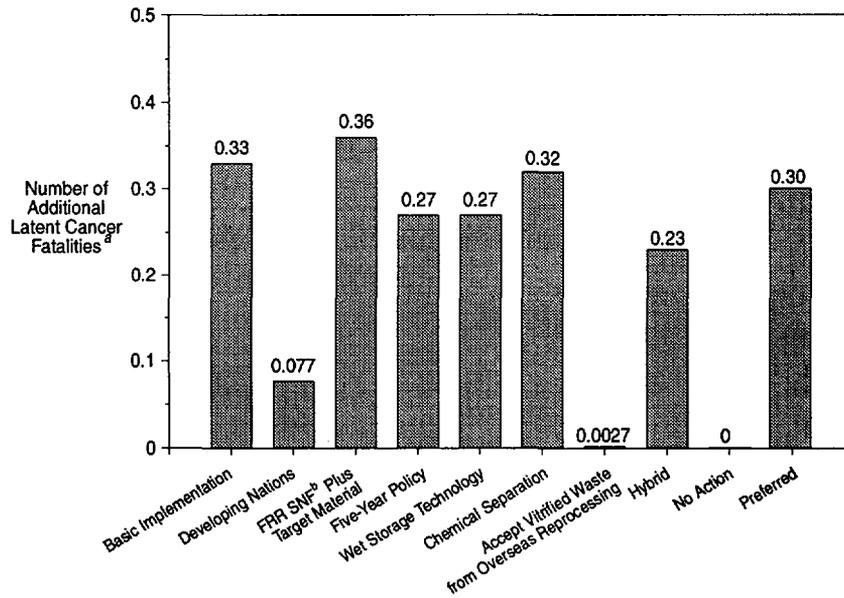
Figure S-11 Maximum Estimated Incident-Free Radiological Population Risk to the General Public Under Each Alternative



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

^b Foreign Research Reactor Spent Nuclear Fuel

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



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

^b Foreign Research Reactor Spent Nuclear Fuel

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

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

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

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

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

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

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

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

S.4.9 Costs

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

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

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

^a Dry/Wet New Storage Facilities

^b Adding target material to any of these scenarios would increase scenario costs by 3 to 4 percent

^c The total cost risk to the United States is less than 1/2 the total cost risk since a large portion of the activities under this alternative would occur overseas

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

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

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

S.5 Overview of the Public Comments and DOE Response

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

S.5.1 The Public Comment Process

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

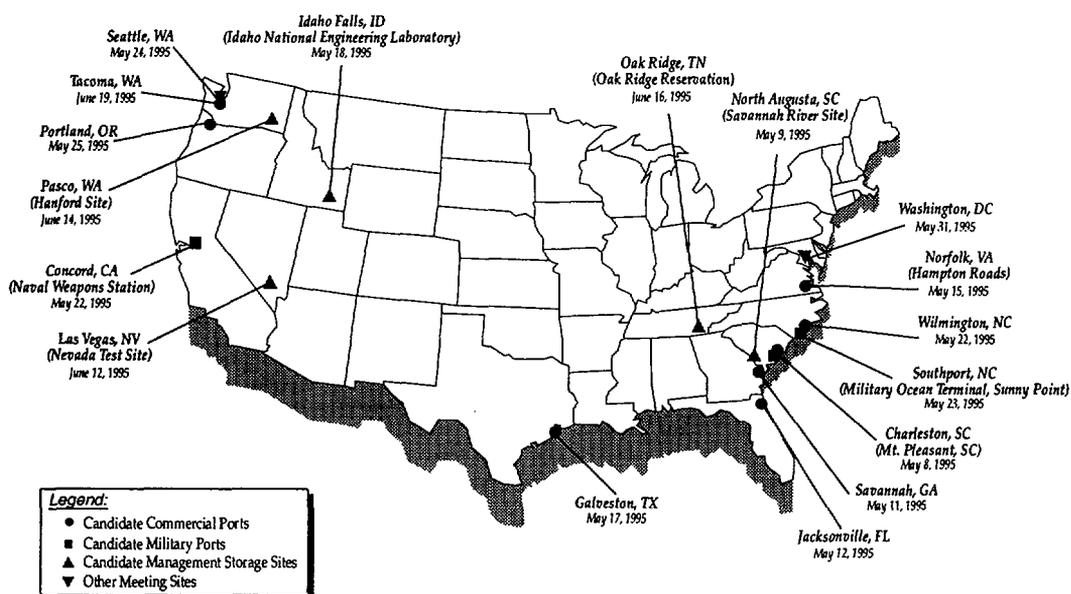


Figure S-14 Public Comment Hearing Locations and Dates

S.5.2 Written Comments

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

S.5.3 Public Hearings

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

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

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

S.5.4 Environmental Protection Agency Rating of EIS

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

S.5.5 Major Issues Raised by Commentors

The public comments addressed a wide range of policy, economic, and technical issues. Of the approximately 6,000 written and oral comments received, few were critical of, or directed against, the analytical methods presented in the Draft EIS. The following is a

summary characterizing the most frequently raised issues and the corresponding summary of DOE's responses. (In each case, a summary of DOE's response is provided in bold text following the summary of the public comment.) DOE's full response to each specific comment and issue are provided in Sections 2 and 3, Volume 3 of the Final EIS.

S.5.5.1 Policy Considerations and Management Alternatives

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

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

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

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

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

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

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

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

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

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

S.5.5.2 Ultimate Disposition

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

S.5.5.3 Transportation and Emergency Response

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

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

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

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

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

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

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

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

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

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

S.5.5.4 Port Selection Criteria and Activities

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

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

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

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

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

S.5.5.5 *Economic Impacts to Candidate Port and Site Communities*

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

S.5.5.6 *Health Effects and Environmental Risks*

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

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

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

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

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

S.5.5.7 Public Involvement Process

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

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

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

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

S.5.6 Availability of the EIS

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

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

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

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

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

S.5.7 Record of Decision

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

S.5.8 DOE Reading Rooms

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

- Department of Energy Reading Rooms -**Public Reading Room for U.S. Department of Energy Headquarters**

Room 1E-190, Forrestal Building
Freedom of Information Reading Room
1000 Independence Avenue, SW
Washington, DC 20585
(202) 586-6020

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Environmental Information Center
1301 Clay Street, Room 700 N
Oakland, CA 94612
(510) 637-1762

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

Front Range Community College Library
3645 W. 112th Avenue, Level B
Westminister, CO 80030
(303) 469-4435

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

Public Reading Room
1776 Science Center Drive
Idaho Falls, ID 83402
(208) 526-9162

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

Government Documents Section
801 South Morgan Street
Chicago, IL 60607
(312) 996-2738

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

87117 Wyoming Boulevard, SE (Kirtland AFB)
Albuquerque, NM 87185
(505) 845-4378

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

Coordination and Information Center
3084 South Highland Drive
P.O. Box 98521
Las Vegas, NV 89106
(702) 295-0731

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

Public Environmental Center
JANTER Building 10845
Hamilton-Cleves Highway
Harrison, OH 44503
(513) 738-0164

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

DOE Public Reading Room
University of South Carolina - Aiken Campus
Grigg-Graniteville Library
2nd Floor
171 University Parkway
Aiken, SC 29801
(803) 641-3320

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

Public Reading Room
55 Jefferson Avenue
Oak Ridge, TN 37831
(615) 576-1216

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

Washington State University Tri-Cities
100 Sprout Road, Room 130 West
Richland, WA 99352
(509) 376-8583

- Other Locations -**Concord Branch Library**

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

George A. Smathers Libraries, Library West

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

Jacksonville Public Library

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

Atlanta Public Library

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

Reese Library

Augusta College
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 Augusta, GA 30904-2200
 (706) 737-1744

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 Savannah, GA 31401
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 Boise, ID 83702
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INEL Oversight Program Library

Idaho Department of Health & Welfare
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 Boise, ID 83706
 (208) 334-0498

Idaho Falls Public Library

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

Pocatello Public Library

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

Twin Falls Public Library

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

Amargosa Valley Community Library

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

Carson City Public Library

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

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

Brunswick County Government Center

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

Pembroke State University Library

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

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 Raleigh, NC 27695-7111
 (919) 515-3364

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Attn: Daniel Horn
 201 Chestnut Street
 Wilmington, NC 28401
 (910) 341-4390

Brantford Price Millar Library

Portland State University
 934 S.W. Harrison
 Portland, OR 97201
 (503) 725-4617

Charleston County Main Library

404 King Street
 Charleston, SC 29403
 (803) 723-1645

South Carolina State Library

1500 Senate Street
 Columbia, SC 29201
 (803) 734-8666

Berkeley County Library

100 Library Street
 Monks Corner, SC 29461
 (803) 722-3550

Otranto Regional Library

2261 Otranto Road
 North Charleston, SC 29418
 (803) 572-4094

Clinton Public Library

118 South Hicks Street
 Clinton, TN 37716
 (615) 457-0519

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500 West Church Avenue
 Knoxville, TN 37902
 (615) 544-5750

Memphis/Shelby County Public Library and Information Center

1850 Peabody Avenue
 Memphis, TN 38104
 (901) 725-8800

Oak Ridge Public Library

Civic Center
 Oak Ridge, TN 37830
 (615) 482-8455

Rosenberg Library

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 Galveston, TX 77550-2296
 (409) 763-2526

Houston Public Library

Attn: Social Sciences
 500 McKinney
 Houston, TX 77002
 (713) 247-2222

Hampton Public Library

4207 Victoria Boulevard
 Hampton, VA 23669
 (804) 727-1154

Newport News Public Library

Grissom Branch
 366 Deshazor Drive
 Newport News, VA 23602
 (804) 886-7896

Kirn Library

301 East City Hall Avenue
 Norfolk, VA 23510
 (804) 441-2429

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Portsmouth Public Library

Main Branch
601 Court Street
Portsmouth, VA 23704
(804) 393-8501

Owen Science & Engineering Library

Washington State University
Pullman, WA 99164-3200
(509) 335-4181

Seattle Public Library

1000 Fourth Avenue
Seattle, WA 96104
(206) 386-4636

Suzallo Library, SM25

University of Washington Libraries
University of Washington
Seattle, WA 98185
(206) 543-9158

Foley Center

Gonzaga University
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Spokane, WA 99258
(509) 328-4220, Extension 3125

Pierce County Library

300 512th Street, East
Tacoma, WA 98446
(206) 536-6500

Tacoma Public Library

1102 Tacoma Avenue South
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(206) 591-5666

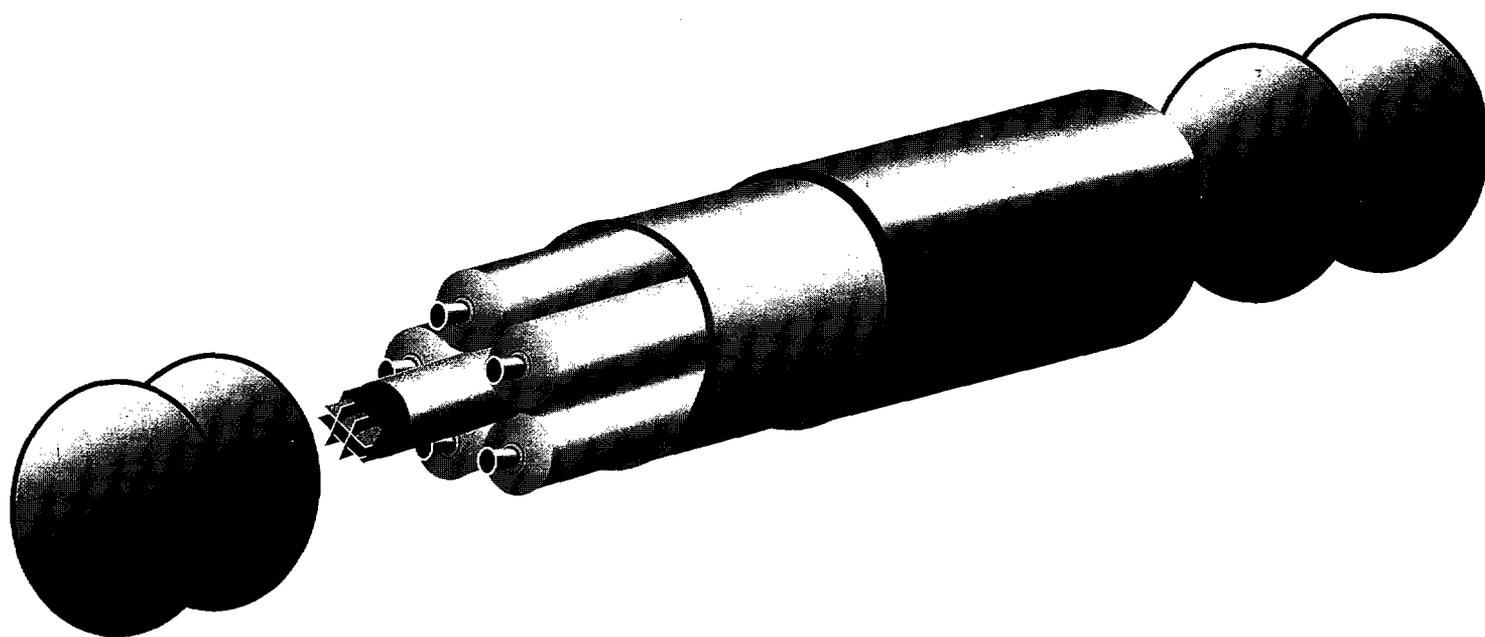
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WASHINGTON, D.C. 20585

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Volume I

Technical Strategy for the Treatment, Packaging, and Disposal of Aluminum-Based Spent Nuclear Fuel

A Report of the Research Reactor Spent Nuclear Fuel Task Team



Prepared For
The Department of Energy
Office of Spent Fuel Management

June 1996

Charles - This is the background information for DOE's program to dispose of AL-based spent nuclear fuel. You should read this first. Jennifer Davis has the files from Kim Cross related to the staff's review of this program.

VERONICA (410)853-0290

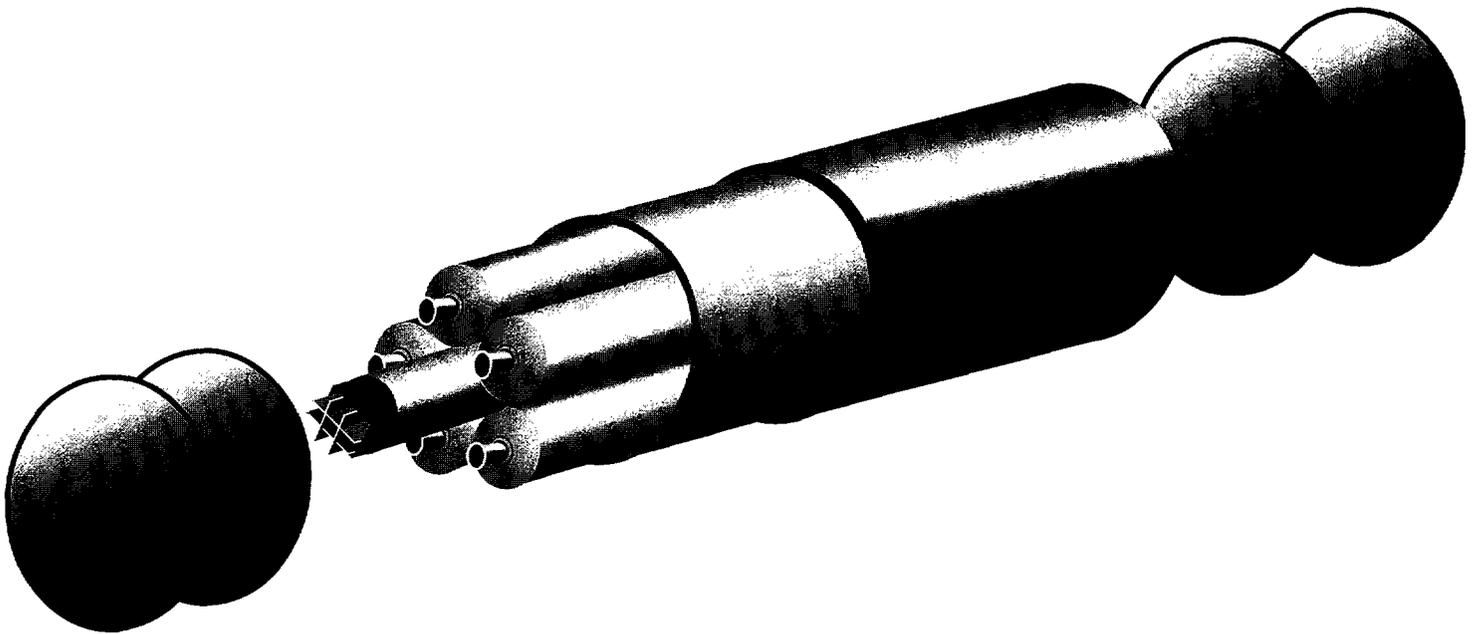
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Volume I

Technical Strategy for the Treatment, Packaging, and Disposal of Aluminum-Based Spent Nuclear Fuel

A Report of the Research Reactor Spent Nuclear Fuel Task Team



Prepared For
The Department of Energy
Office of Spent Fuel Management

June 1996

Preface

This document, *Technical Strategy for the Treatment, Packaging, and Disposal of Aluminum-Based Spent Nuclear Fuel*, is a report to the U.S. Department of Energy (DOE) by the Research Reactor Spent Nuclear Fuel Task Team.

During the six-month period from November 1995 through April 1996, the Task Team examined the wide-ranging technical issues attendant to achieving safe and cost-effective disposal of the aluminum-based spent nuclear fuel under DOE's jurisdiction. This fuel is from research and test reactors, both U.S. and foreign; some is already in temporary wet storage at DOE's Savannah River Site, and the balance will be sent there over the next two decades.

Disposal of aluminum-based fuel raises issues which are different, and in some respects more challenging, than those presented by other types of DOE and commercial spent nuclear fuels. Given the expected unavailability of fuel processing capability beyond current operational commitments, alternative means of rendering this fuel safe and stable for ultimate disposal are required. DOE convened the Task Team to identify and evaluate treatment/packaging alternatives, and to recommend a technical strategy for the selection, development and implementation of one or more alternatives.

This report, and its underlying evaluations, are primarily technical in nature. While DOE's final decisions on the path forward for the aluminum-based fuel must properly take into account the full spectrum of budget, policy and other institutional considerations, the Task Team was charged with an evaluation of the issue on technical grounds, and the findings and recommendations herein reflect its consensus judgments as to the technically preferred course of action.

Achieving a final and satisfactory resolution of this multifaceted matter will be a continuing process of assessment, development, adjustment and convergence. This report offers a path forward, an important next step in that process.

Cover Illustration: Artist's rendition of Repository Co-Disposal Waste Package, with the spent nuclear fuel canister highlighted in blue (see Figure 1.3-1).

■ The Research Reactor Spent Nuclear Fuel Task Team

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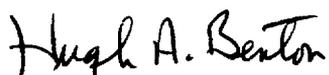
Core Team Concurrence



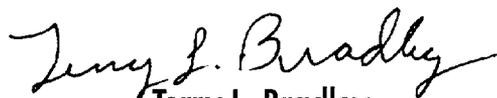
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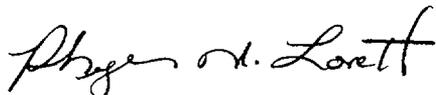
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1. The Assessment - An Overview

In November 1995, the Department of Energy (DOE) Office of Spent Fuel Management established the Research Reactor Spent Nuclear Fuel Task Team to assist in developing a technical strategy for interim management and ultimate disposition of the foreign and domestic aluminum-based research reactor spent nuclear fuel (SNF) in DOE's jurisdiction, including both current inventory and expected receipts. The Team's work is documented in this two-volume report. Volume I (this volume) provides a technical synopsis of the fuel in question and the issues involved, and summarizes the Team's evaluations, findings, and recommendations to DOE. Volume II is a compendium of supporting technical information.

The following is a brief overview of this work.

■ 1.1 Background

Within DOE, two organizations share the overall responsibility for assuring the safe disposal of the nation's SNF inventory. The Office of Civilian Radioactive Waste Management (OCRWM) is responsible for SNF and high-level radioactive waste (HLW) disposal, including the development and licensing of a geologic repository. The Office of Environmental Management (EM) is responsible for the interim management and preparation for disposal of the non-commercial (e.g., DOE-owned) SNF. The latter primarily includes SNF generated within DOE's nuclear materials (including nuclear weapons materials) and research programs. EM's Office of Spent Fuel Management (EM-67) establishes the methods to be employed in the treatment, packaging, storage, and preparation for disposal of the SNF, including both current inventory and expected receipts.

This aluminum-based SNF is from research and test reactors in the United States, or from foreign research reactors (FRR).^a All is either presently stored at the Savannah River Site (SRS) or is planned for future storage there after shipment from other U.S. locations, including DOE sites, or the FRR sites.

While research reactor fuel is a relatively small part of the DOE's SNF inventory (see Figure 2.1-1) – and very small when contrasted to the large quantities of commercial power reactor fuel requiring disposal – it is quite significant in terms of technical challenge. About three quarters (74%) by volume is in the form of high-enriched uranium (HEU)^b, introducing criticality control issues not present for commercial fuel disposal. The fuel is constructed of aluminum (either as cladding or as an uranium-aluminum alloy fuel material) and is therefore more

^a For purposes of this report all of this material will be considered "research reactor SNF."

^b HEU is defined as material for which >20% of the uranium is U-235. Low-enriched uranium (LEU) contains ≤20% U-235.

vulnerable to corrosion than commercial SNF. There are several different types of research reactor fuel, of various size, shape, material, and structural configuration, with attendant differences in handling, packaging, and treatment requirements.

Establishing a success-oriented course of action for the management and disposal of the research reactor fuel is a matter of high priority to DOE. Disposal in the geologic repository is not anticipated until 2015, and perhaps later, necessitating provisions for the safe interim management of this material for a potentially long time. The continued use of existing wet storage facilities at SRS that are not adequate or cost-effective for extended storage of SNF cannot be relied upon for the long term.

The expected unavailability of SRS processing capability beyond about 2002 is another important issue. This loss of processing capability creates an immediate need to identify those select fuels having unique safety, technical, or other characteristics that present difficult or expensive disposal options. For these few fuels, processing within the existing SRS canyons while they are operating may be the most practical solution. For the remainder, the identification of new treatment and interim management capability is essential.

Finally, safe handling and disposal of foreign research reactor fuel is in the United States' national interest, consonant with U.S. policy and concerns regarding nuclear materials safeguards.¹ It is important for DOE to demonstrate to national and local stakeholders that DOE has a well conceived, viable path forward for this material.

■ 1.2 The Task Team

The DOE Office of Spent Fuel Management chartered the Research Reactor Spent Nuclear Fuel Task Team to assist in developing a technical strategy for interim management and ultimate disposition of the research reactor SNF. Team members were drawn from the DOE complex and outside support organizations, providing a broad base of experience and capability germane to this problem. They were organized into a small core unit, a group of specialists in supporting roles, and a DOE steering committee.

The Team's primary task was to evaluate and compare alternative treatment and packaging technologies which could be utilized in place of conventional processing to achieve safe and cost effective interim storage and ultimate disposal of the SNF. The Team was asked to produce a recommended course of action, leading to final technology selection(s), development and (if possible) implementation by the year 2000.^c

Eleven technologies were initially identified for potential use in the treatment of research reactor SNF (see Table 1.2-1). Several are applications of existing technologies while others may be considered "new" in their application to the treatment of SNF. These technologies

^c The Team's assignment is consistent with DOE's Record of Decision (ROD) pursuant to the Environmental Impact Statement on the Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Spent Nuclear Fuel.

Table 1.2-1. Treatment Technologies for Aluminum-Based Spent Nuclear Fuel

Direct Disposal - The SNF would be placed into small waste packages, ready for direct disposal, with fuel quantities limited to satisfy repository criticality requirements.

Direct Co-Disposal - The SNF canisters would be disposed by placement in repository waste packages which contain HLW glass logs. Fuel quantities may be limited to satisfy repository criticality requirements.

Can-in-Canister - A critically safe quantity of SNF would be placed in a can. This can is placed in a canister into which HLW glass is poured to form a solidified unit.

Press and Dilute/Poison - To minimize volume, the SNF would be mechanically compressed and either diluted with depleted uranium or mixed with a neutron poison.

Chop and Dilute/Poison - The SNF would be chopped into small pieces and diluted with depleted uranium or mixed with a neutron poison.

Melt and Dilute - The SNF would be melted and diluted with depleted uranium.

Plasma Arc Treatment - The SNF would be placed directly into a plasma centrifugal furnace with depleted uranium and neutron absorbers, where it would be melted and converted into a HLW ceramic waste form.

Glass Material Oxidation and Dissolution System - The SNF would be placed in a glass melt furnace where it is oxidized by lead dioxide and converted into a HLW glass waste form.

Dissolve and Vitrify - The SNF would be dissolved and mixed with depleted uranium to dilute the HEU to LEU. The mixture is then fed into a vitrification plant for conversion to a HLW glass waste form.

Electrometallurgical Treatment - The SNF would be melted with silicon and electrorefined. The bulk of the aluminum would be electrolytically removed for disposal as low-level waste; the residual aluminum, actinides, and fission products would be vitrified. Pure uranium would be recovered.

Chloride Volatility - The SNF would be reacted at high temperatures with chlorine gas and all of the materials converted to a volatile gas. The uranium, actinides, and fission products would be separated from each other by cooling and distillation.

differ in terms of technical maturity, facilities required, costs, and waste forms produced. Some are relatively simple in their technical approach and can be expected to be less costly to implement than others. Other considerations, however, such as repository waste form acceptability and flexibility to accommodate a wider range of materials, may favor the more complex technologies. The Team employed a comparative scenario (processing using existing SRS facilities followed by co-disposal) for comparison with the alternative technologies.

During its six-month term, the Team conducted a series of parallel investigations, compiling technical information on the SNF in question, on the candidate treatment technologies, on repository requirements, and on storage, handling, and packaging systems. In each area, experts and advocates were engaged to provide the fullest possible understanding of the matter at hand. The Team then conducted a comparative evaluation of the treatment technologies, considering such factors as technical suitability, cost, schedule, and technical maturity.

■ 1.3 Findings

From its evaluation and comparison of technologies, the Team derived the following broad set of findings:

- Numerous viable options exist for the safe, cost-effective treatment, packaging, storage, and disposal of research reactor SNF. DOE's timetable can be met.
- A determination of repository waste form acceptance is essential for all of the technologies. While all of the waste forms under consideration are expected to be technically satisfactory, some may be more difficult to qualify than others in that they are significantly different than commercial SNF or HLW.
- (As a packaging strategy, co-disposing any of the waste forms with packaged vitrified HLW appears technically sound and cost-effective.^d (See Figure 1.3-1.)
- For a few of the research reactor fuel types (a small part of the inventory), processing using existing SRS facilities is the most practical and cost-effective approach. These small quantities could be accommodated easily at SRS while the canyons are still available for other uses.
- For all technologies considered, a new facility for receipt, handling, and packaging of fuel appears cost beneficial in that it would permit early closure of the existing SRS wet storage basins. (See Figure 1.3-2.)

^d Vitrified HLW is the "glass log" waste form created by the Defense Waste Processing Facility (DWPF) at SRS, or comparable processes.

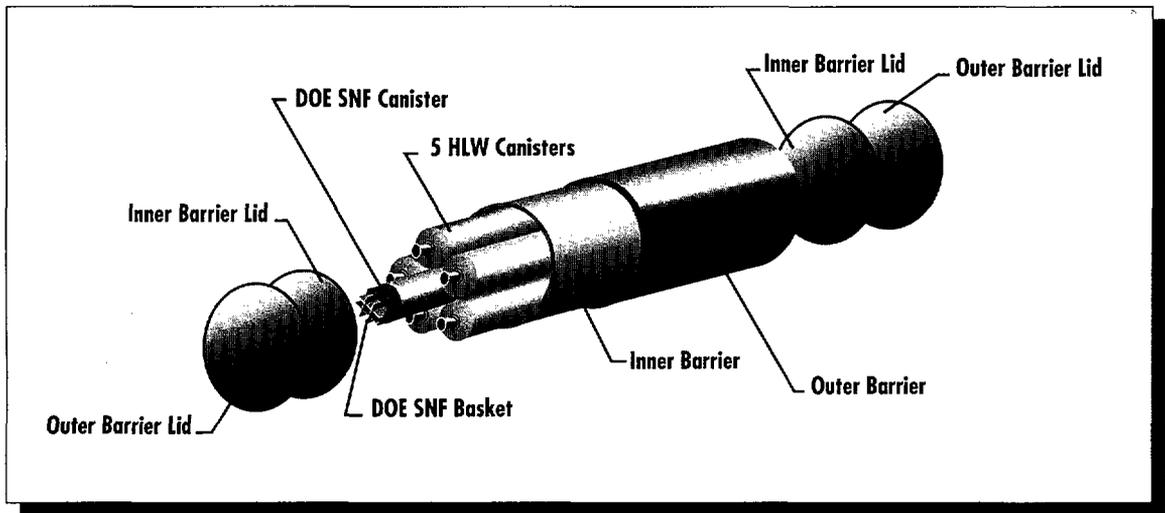


Figure 1.3-1. Co-Disposal Package — A Packaging Strategy Attractive for All Options

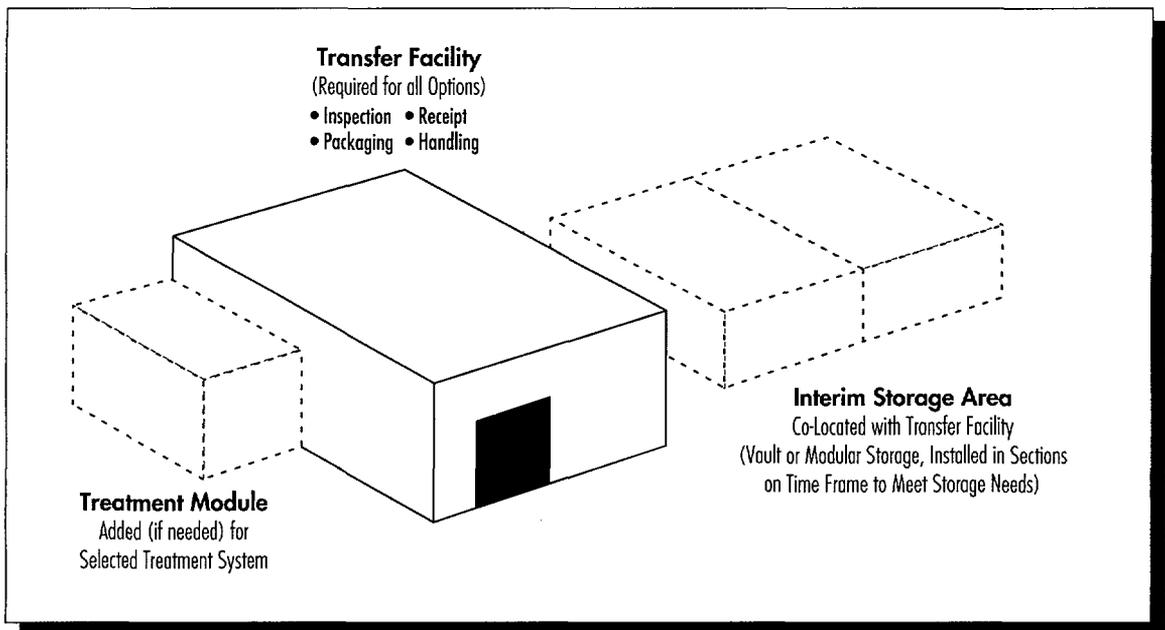


Figure 1.3-2. Spent Nuclear Fuel Dry Transfer Facility (Needed for all Options)

■ 1.4 Recommendations

Based on these findings, the Team recommends the following course of action.

- DOE should proceed with the parallel development of at least two treatment technologies. The recommended choices are:
 - **Direct Disposal** (co-disposal packages) as the primary approach.
 - A dilution option, **Press and Dilute** or **Melt and Dilute** (as selected by further evaluation), as the backup.The final approach may require one or both of these in concert with selective processing.
- DOE should utilize the existing SRS processing capability for those few fuel types for which processing clearly is the most cost effective and timely treatment method.
- DOE should retain an advanced technology option as a secondary and diverse backup, should one be needed. Electrometallurgical Treatment was evaluated as a potentially strong technical candidate. Its development is well along and will continue without additional funding from the spent fuel program, while the direct disposal and the dilution technologies are being developed.
- DOE should begin immediately to work with the Nuclear Regulatory Commission (NRC) and other regulatory authorities to reach agreement on SNF disposal requirements, and particularly on HEU waste forms.
- DOE should move ahead with the planning, funding, and design for an SNF transfer facility. It should be designed to accommodate a treatment module based on technology selection(s) made later.
- To achieve the schedule desired by DOE, a strong, focused and results-oriented effort is required. It is recommended that a project approach be employed.

The following sections of this Volume I provide further detail regarding the Team's evaluations, findings, and recommendations.

2. The Problem - Scope, Dimensions and Technical Issues

2.1 Aluminum-Based SNF: An Overview

Aluminum-based research reactor fuel is a small part (less than 1% by volume) of the total inventory of SNF and HLW that is to be disposed in a geologic repository (Figure 2.1-1). This small amount, however, is a significant challenge because it includes a number of fuel types having differing geometries, materials, and characteristics.

Aluminum-based SNF is currently stored under water at all sites, but continued long-term wet storage is not desirable. Aluminum is more susceptible to corrosion in water than the materials used in other fuel types, and several of the aluminum-based fuel types have degraded in wet storage. Improvements in water chemistry control and canisterization have greatly reduced this ongoing degradation, such that there are no near term health and safety concerns for the workers and the public.^c However, the underlying concerns regarding the adequacy of wet storage remain.

Much of the aluminum-based research reactor SNF is significantly higher in enrichment than commercial fuel, a distinction which is potentially important for repository disposal. Commercial fuels generally have enrichment values of about 2%. Many research reactor fuels

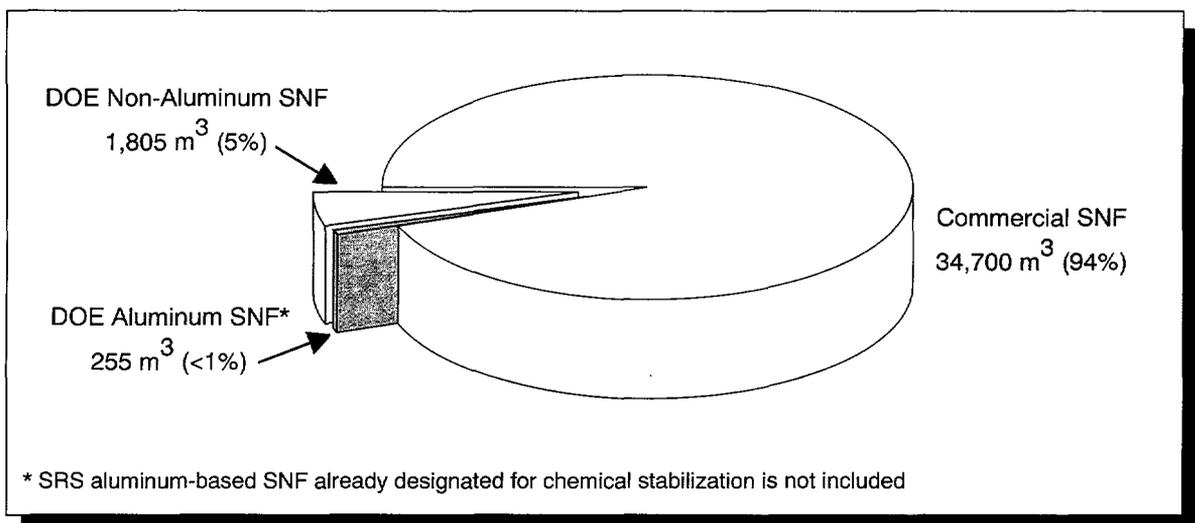


Figure 2.1-1 Total SNF Volume for Repository Disposal through 2035 – Commercial/DOE

^c For several of the existing fuels, the concern for further degradation was such that DOE has already decided to process these fuels in the Savannah River canyons. For additional information, please refer to the Supplemental Record of Decision (issued 2/96) on the Interim Management of Nuclear Materials at the Savannah River Site.

have enrichments that exceed 90%, and even the lower enriched ones have enrichments of about 20%. Higher enrichments pose potential concerns regarding criticality and diversion that must be considered carefully in the evaluation of disposal options.

Some of the fuel types included in this evaluation (about 6% of the total by volume) may not be suitable for direct disposal, per present regulatory interpretation, without treatment. These are uranium metal fuels, uranium-thorium metal fuels, particulate fuels, and failed and sectioned fuels. Also, some target material fuels are normally shipped in forms not well suited to disposal (powders) and a determination needs to be made whether they can be shipped by the reactor operators in a more suitable form.

Less than 10% of the research reactor SNF is currently stored at the SRS. Not counting that which is slated for processing, SRS currently stores 16.7 m³ (33.5 metric tons heavy metal [MTHM]) of aluminum-based SNF. This quantity is expected to increase to 255 m³ (62.4 MTHM) by 2035 due to redistribution of SNF from other DOE sites, including the Idaho National Engineering Laboratory (INEL), and to SNF receipts from domestic research reactors and from foreign research reactors (Figure 2.1-2).^f Figures 2.1-3 and 2.1-4 depict projections of

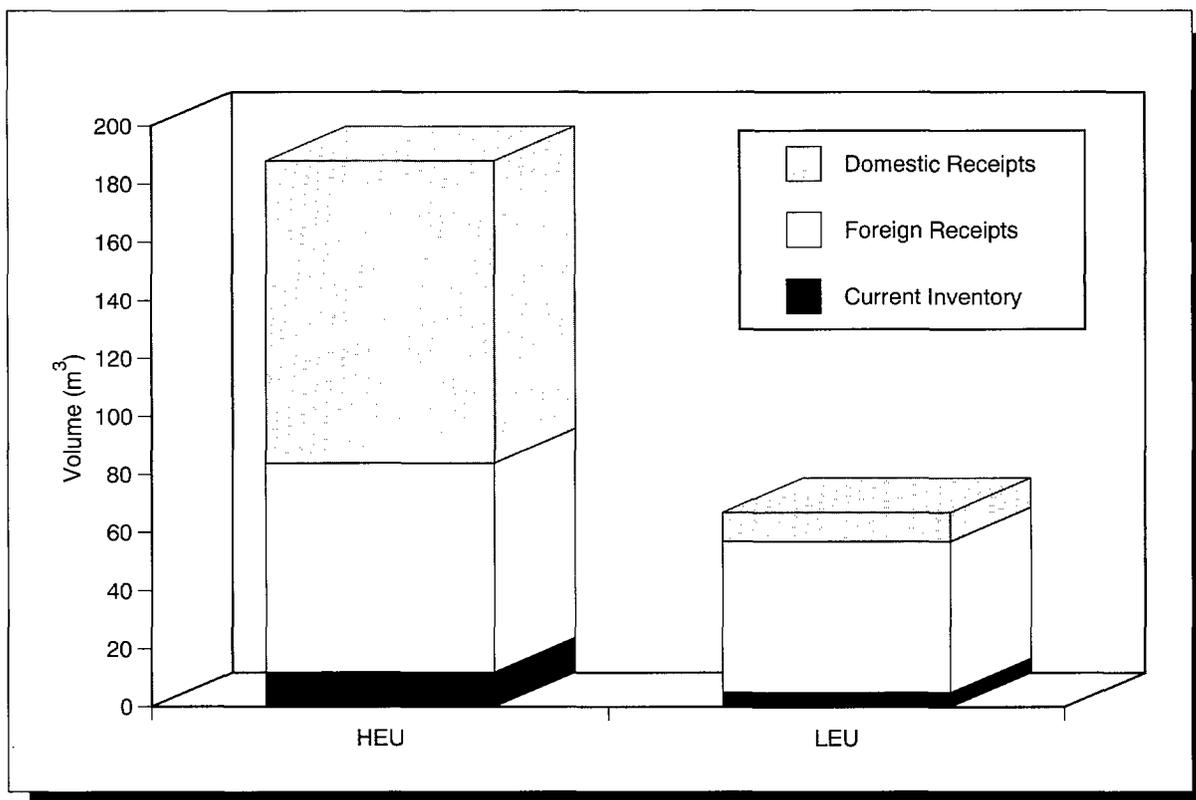


Figure 2.1-2 Aluminum-based HEU and LEU SNF to be Managed at SRS

^f Various parameters are commonly used to quantify SNF, including volume, number of assemblies, total weight and weight of heavy metals. For consistency, units of volume will be used in this report, unless another parameter is more meaningful within a particular discussion.

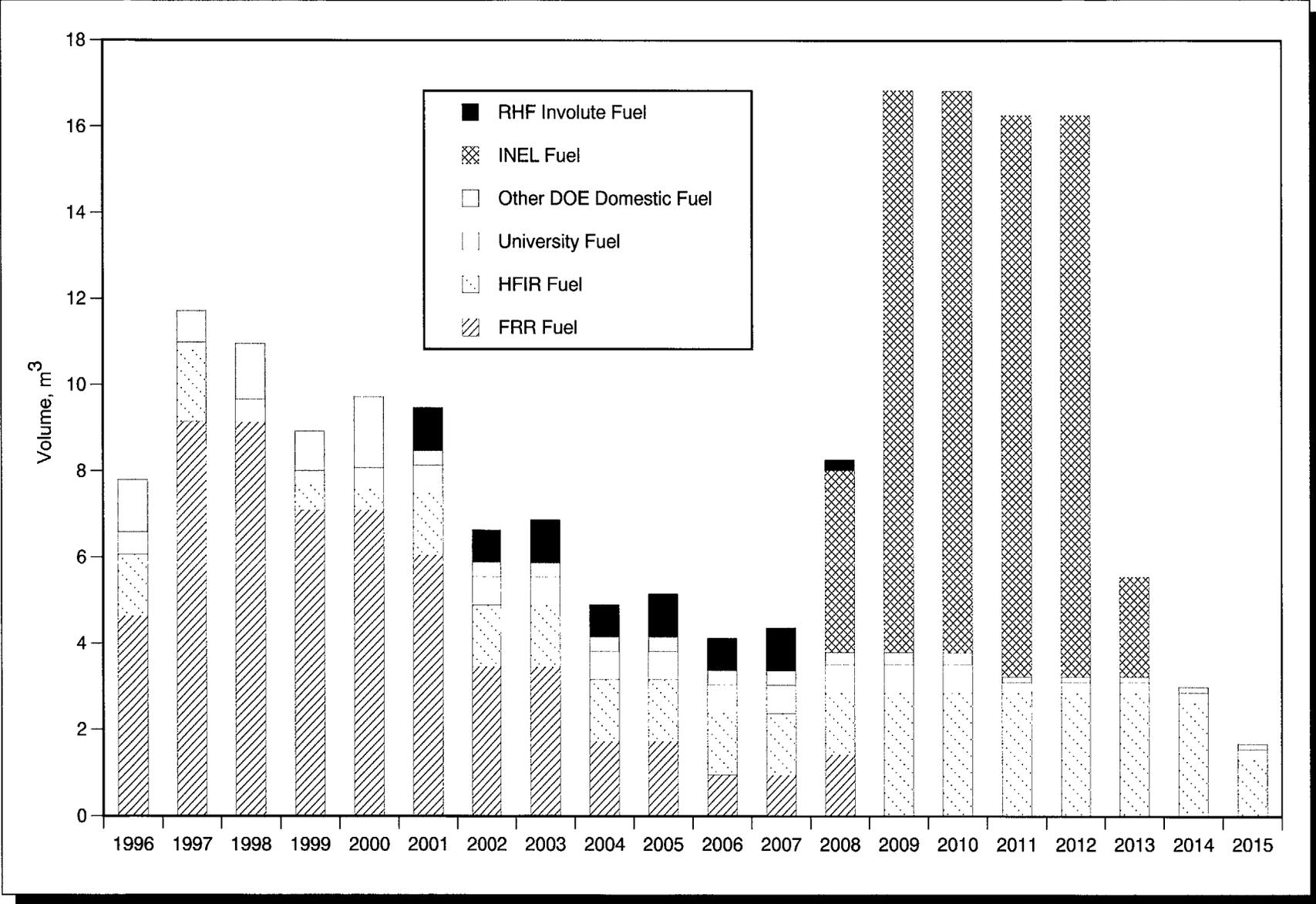


Figure 2.1-3 HEU SNF Projected Receipts, by Source

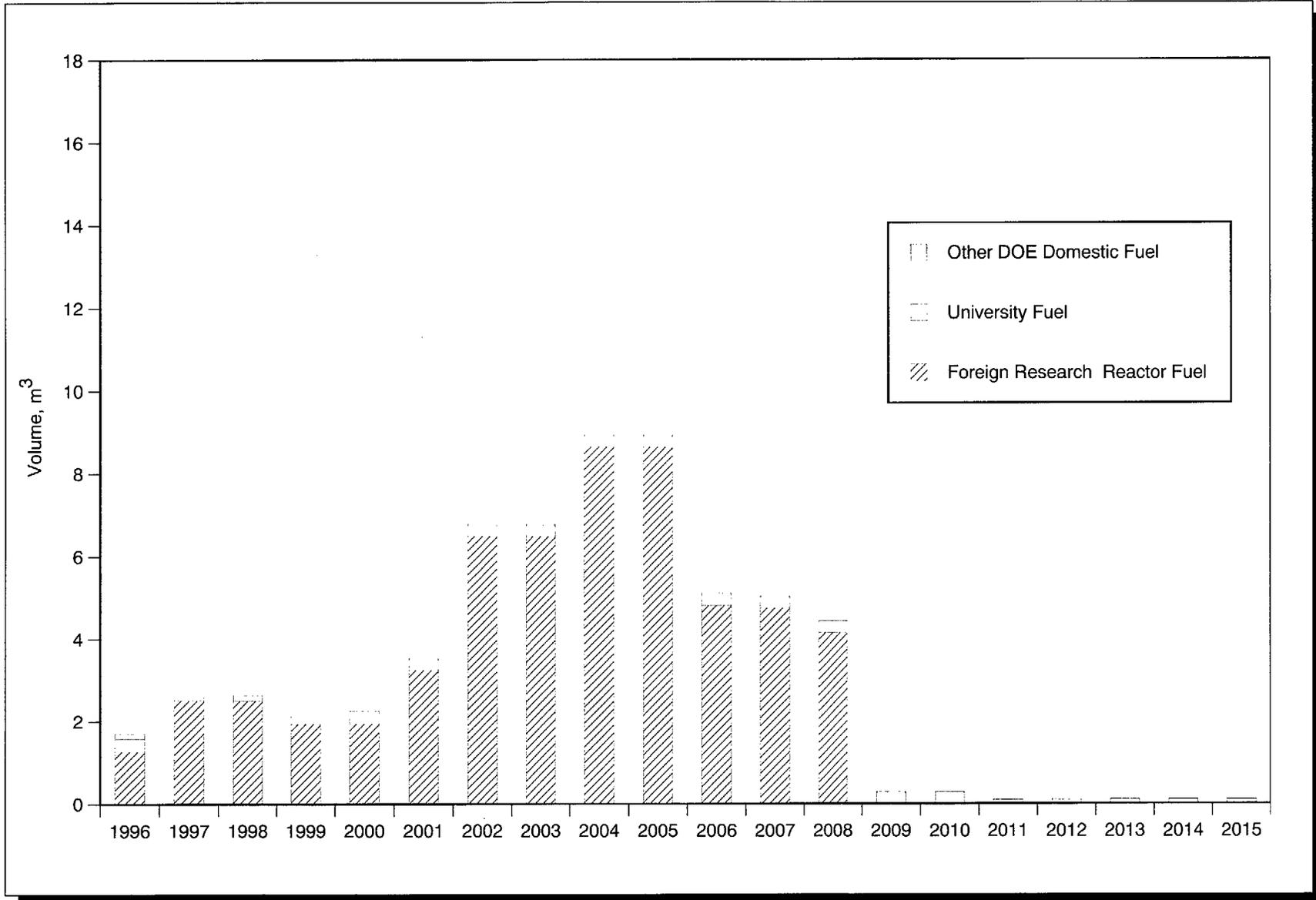


Figure 2.1-4 LEU SNF Projected Receipts, by Source

the annual SRS receipt rates of high and low-enriched research reactor SNF; these were used by the Team to estimate the treatment and handling requirements for the candidate treatment technologies.

Fact sheets for the major research reactor fuel types are included on pages 13-19. More detailed discussion of SNF inventories and projected receipts can be found in Appendix A.

■ 2.2 Issues

Among the many technical and policy issues which DOE must address in establishing a successful path forward for the safe and economical disposal of the SNF in question, the following are particularly significant:

Adequacy of Existing Storage Facilities

Although DOE has taken positive steps to improve the safety of existing SNF storage at SRS, wet storage is not suitable over the long term. In particular:

- Long-term underwater storage of aluminum fuels is not desirable because of the potential for corrosion, even with well-maintained water chemistry.
- Wet storage costs are high. Operations and maintenance (O&M) costs are substantially greater on a unit basis for wet storage than for dry, and some of the existing wet storage facilities do not meet current requirements and would have to be upgraded for long-term use.
- Available storage space and cask handling capabilities are limited and may not meet future needs.

Repository Availability

The repository availability date, currently estimated at 2015 for DOE SNF, is uncertain. Therefore, interim storage must be suitable for an extended and as yet indeterminate period of time.

Repository Acceptance Criteria

Repository acceptance criteria are not yet established, and are vulnerable to change in the years prior to repository availability. The recommended path forward must recognize and accommodate this uncertainty. Of particular importance are:

- Materials requirements, due to the susceptibility of aluminum to corrosion.
- HEU disposal requirements, considering nuclear criticality and diversion potential.

Cost

The recommended course of action must be compatible with practical limitations on DOE funding, both near term and long term. Furthermore, predictability of costs for SNF treatment and disposal options is limited; recommendations must take cost uncertainty into account.

Proliferation

The recommended technical strategy must be consistent with U.S. policy with respect to nonproliferation of nuclear weapons materials.

Availability of Processing Capability

Processing capability at SRS is currently projected to be available only until existing commitments are met; that is, until about 2002.⁸ Other SNF for which processing is necessary or desirable must be identified and authorized in that time frame.

Decision Timing

Based on plans to phase out processing capability at SRS, DOE must have in place by the year 2000 a firm plan for disposition of the SNF addressed in this study.

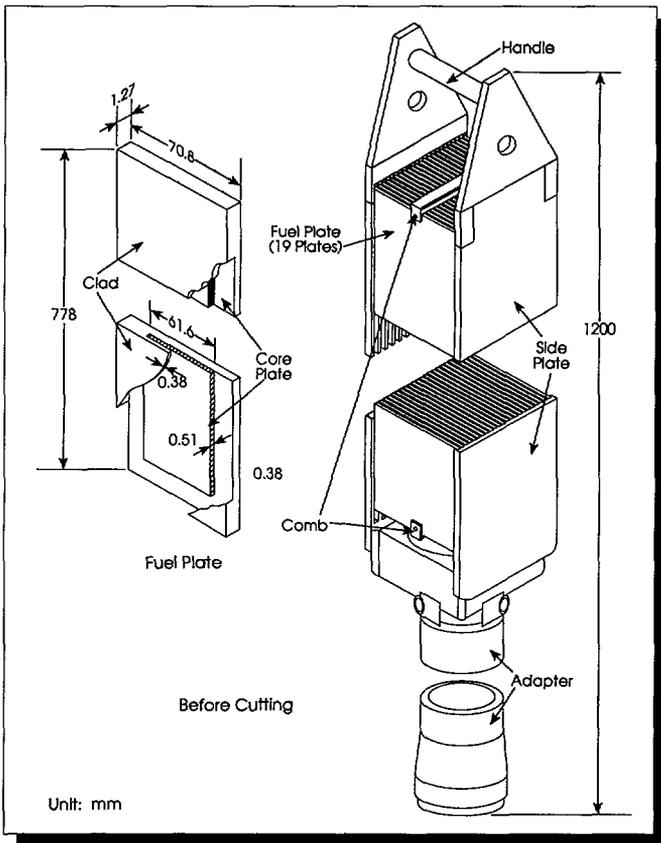
⁸ For additional information on existing commitments made by DOE to the Defense Nuclear Facilities Safety Board (DNFSB), please refer to the "Defense Nuclear Facilities Safety Board Recommendation 94-1 Implementation Plan," U.S. Department of Energy, February 1995.

Fuel Facts

Materials and Test Reactor (MTR) Fuel

Fuel Description

MTR fuel elements are used in the majority of research reactors. These fuel elements exist in a variety of shapes of roughly the same size, and typically the reactors operate at thermal powers from 1 megawatt (MW) to 50 MW. MTR spent fuel elements make up 71% of the volume of aluminum SNF that will be managed at the Savannah River Site (SRS) by 2015.



Physical Characteristics

- Dimensions - 1.2 meters long (0.8 meter with ends cut) by <8 cm square
- Materials - Uranium-aluminum, uranium oxide-aluminum, or uranium silicide-aluminum alloy, clad with aluminum
- Uranium - Typically less than 2 kg of uranium per element
- Enrichment - U-235 enrichment varies from just below 20% to as high as 93%
- Construction - Rectangular or cylindrical cross-section containing up to 25 flat or curved plates

Number of Elements

- HEU - approx. 22,000 by 2015
- LEU - approx. 9,200 by 2015

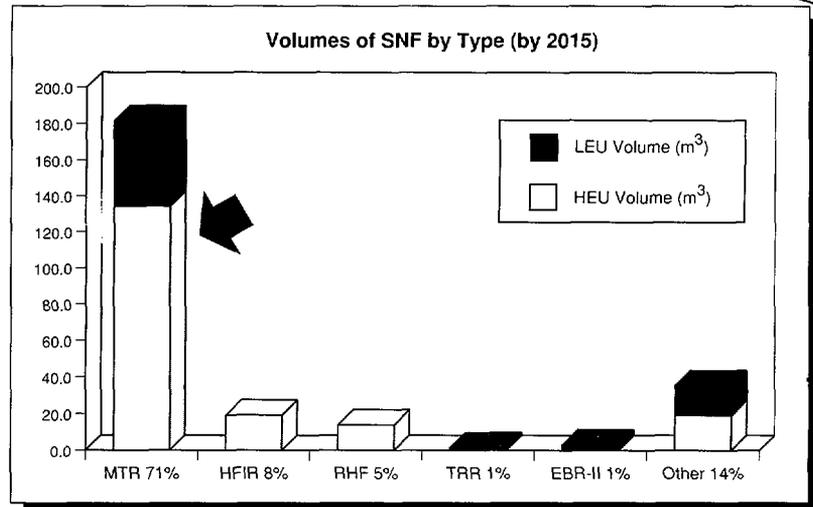
Sources and Condition

- University research reactors in U.S., foreign research reactors
- Fuel is intact with good cladding integrity

Inventory

Current HEU inventory at SRS is 6.4 m³, but will grow to 134 m³ by 2015. Current LEU inventory is 0.8 m³, but will grow to 47 m³, by 2015.

Receipts of HEU MTR fuel are expected to decline from 2002 - 2008 as foreign reactors convert to LEU fuels; LEU receipts will increase for the same reason.

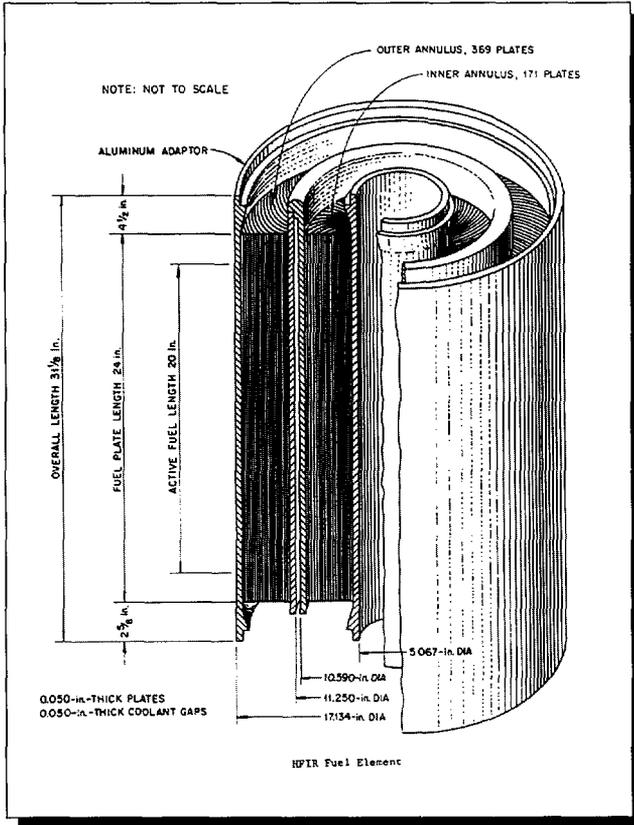


Fuel Facts

High Flux Isotope Reactor (HFIR) Fuel

Fuel Description

HFIR uses a dual annulus, involute fuel plate design. Only one element is used in the reactor at a time, typically operating at a thermal power of 85 MW. HFIR spent fuel elements make up about 8% of the volume of aluminum SNF that will be managed at the Savannah River Site (SRS) by 2015.



Physical Characteristics

- Dimensions - 0.80 meters long with an outside diameter of 43 cm
- Materials - Uranium oxide-aluminum matrix, clad with aluminum
- Uranium - 10 kg of uranium per element (total element weight is 136 kg)
- Enrichment - U-235 enrichment of 93%
- Construction - Primarily aluminum with 171 involute fuel plates in the inner annulus, and 369 involute fuel plates in the outer annulus

Number of Elements

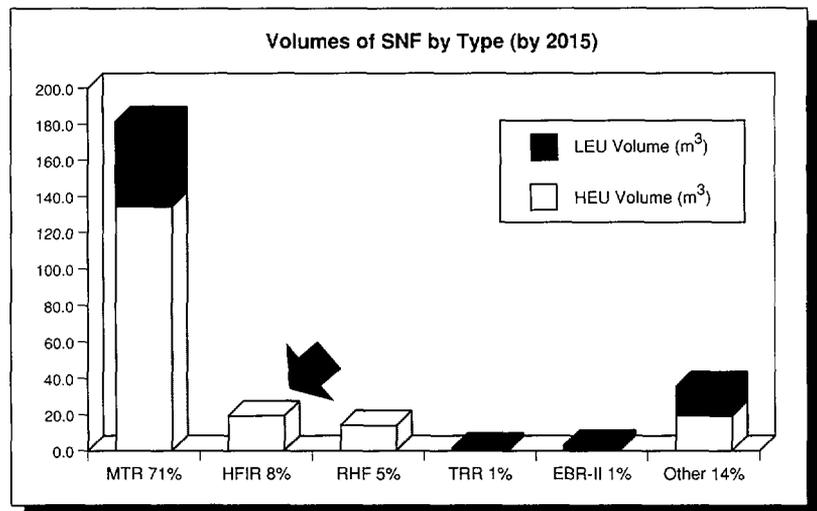
- 284 HEU elements expected by 2015 (one element discharged from the reactor per month)

Sources and Condition

- High Flux Isotope Reactor at Oak Ridge
- Fuel is intact with good cladding integrity

Inventory

Currently there are no HFIR spent fuel elements at SRS, but by 2015 they will make up approximately 19.3 m³ of the HEU SNF. Shipments of HFIR fuel from the storage basin at Oak Ridge are expected to begin in 1996 and continue at a constant rate through 2009, and at double that rate through 2015.

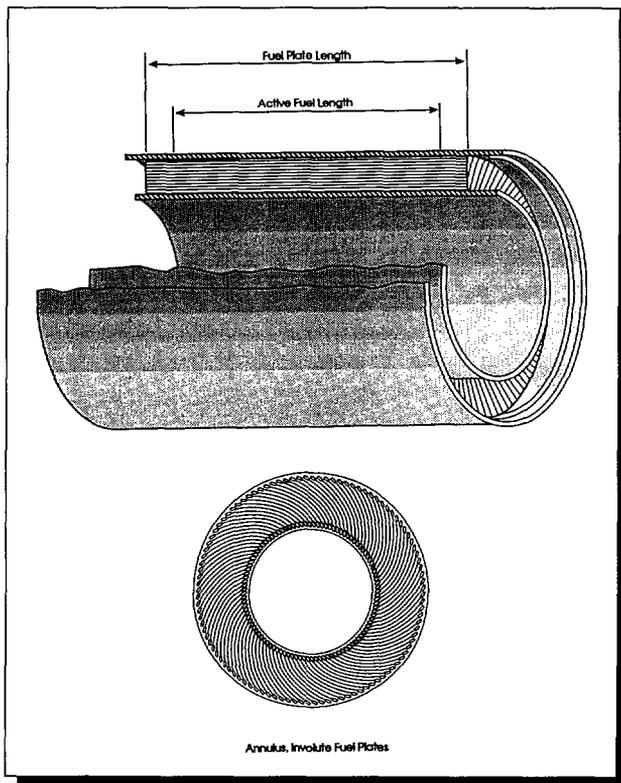


Fuel Facts

Reactor a-Haut Flux (RHF) Fuel

Fuel Description

RHF uses an annulus, involute fuel plate design. Only one element is used in the reactor at a time, typically operating at a thermal power of nearly 60 MW. RHF spent fuel elements make up more than 5% of the volume of aluminum SNF that will be managed at the Savannah River Site by 2015.



Physical Characteristics

- Dimensions - 0.97 meters long with an outside diameter of 41 cm and inside diameter of 27 cm
- Materials - Uranium-aluminum alloy, clad with aluminum
- Uranium - Up to 9.2 kg of uranium per element (total element weight of 100 kg when cut)
- Enrichment - U-235 enrichment as high as 93%
- Construction - Up to 280 involute fuel plates in the annular space between two aluminum tubes

Number of Elements

- 86 HEU elements expected by 2015

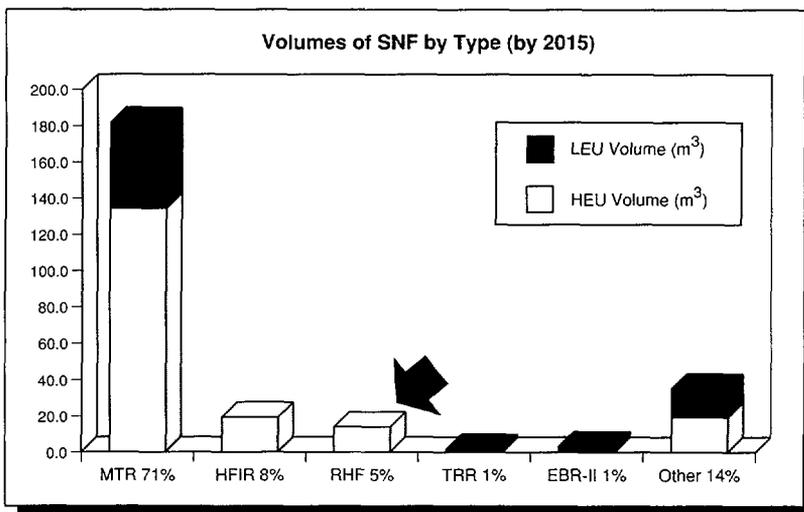
Sources and Condition

- RHF is a French research reactor
- Previous receipts have been in good condition

Inventory

RHF fuel constitutes about 0.5 m³ of the current HEU SNF at SRS, and will be approximately 14.0 m³ by 2015.

Shipments of RHF fuel from France to the SRS are expected to resume between 2001 and 2008.

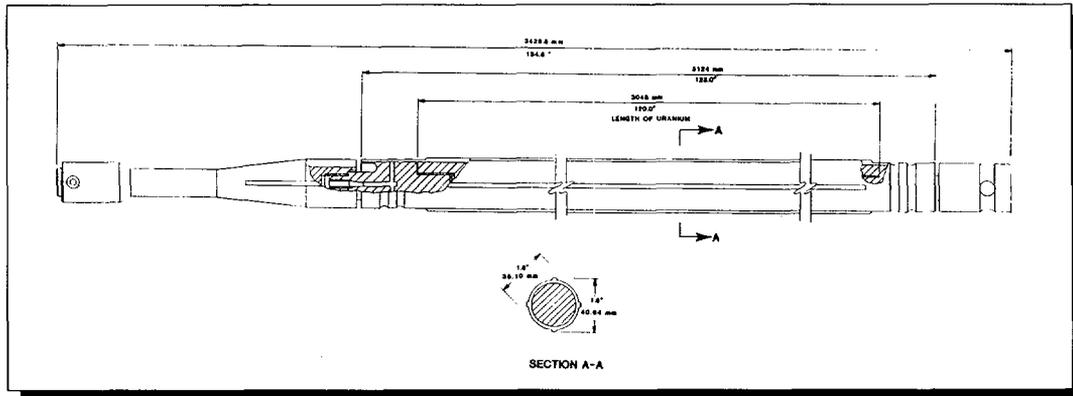


Fuel Facts

Taiwanese Research Reactor (TRR) Fuel

Fuel Description

TRR fuel is configured in long natural uranium fuel rods. Due to their poor condition they have been enclosed in canisters for storage. TRR spent fuel comprises less than 1% of the volume of aluminum SNF that will be managed at the Savannah River Site (SRS) by 2015. Condition of the TRR fuel warrants special consideration.



Physical Characteristics

- Dimensions - 3.0 meters long by 13 cm diameter canisters containing fuel rods
- Materials - Uranium metal clad with aluminum
- Uranium - 140 kg total uranium per canister
- Enrichment - Natural or depleted
- Construction - Large canisters containing uranium metal rods clad in aluminum

Number of Elements

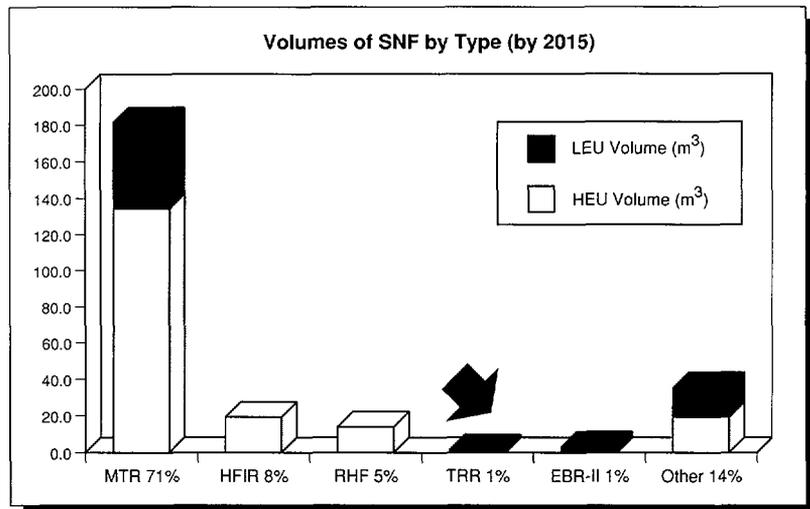
- 143 canisters currently in inventory at the Receiving Basin for Offsite Fuels (no additional receipts planned)

Sources and Condition

- Taiwanese Research Reactor
- 81 of the 143 canisters are degraded, and the decision has already been made to process them. The remaining 62 canisters are intact; the fuel inside is degraded but its precise condition is not known.

Inventory

There are 143 canisters of TRR spent fuel rods at SRS. The 62 canisters not already designated for processing contain 1.5 m³ of aluminum LEU SNF.



Fuel Facts

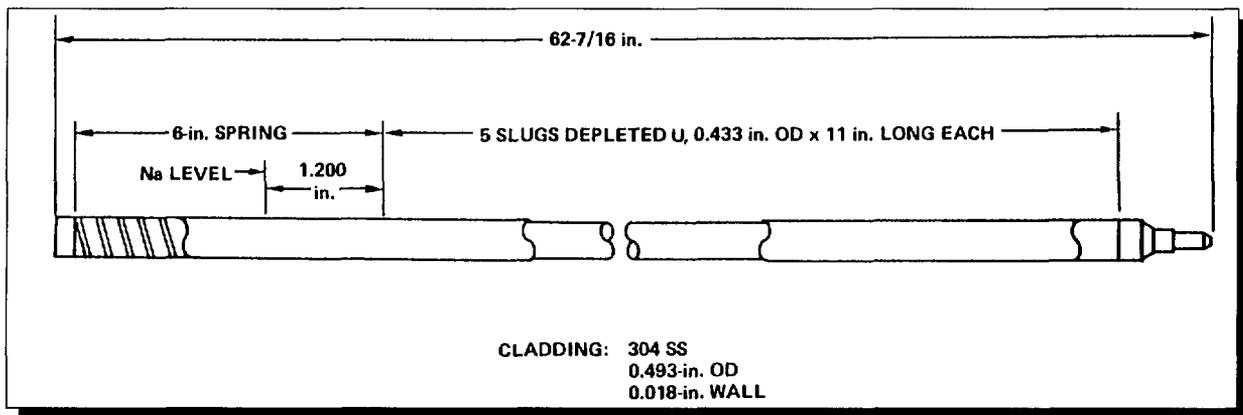
Experimental Breeder Reactor (EBR-II) Blanket Fuel

Fuel Description

EBR-II blanket fuel rods are enclosed in long aluminum canisters. EBR-II fuel is about 1% of the volume of aluminum SNF that will be managed at the Savannah River Site (SRS) by 2015.

Physical Characteristics

- Dimensions - 4.2 meters long by 12 cm diameter aluminum canisters containing blanket fuel rods
- Materials - Declad uranium metal
- Uranium - 280 kg total uranium per canister
- Enrichment - Natural or depleted
- Construction - Large aluminum canisters containing declad uranium metal slugs



Number of Elements

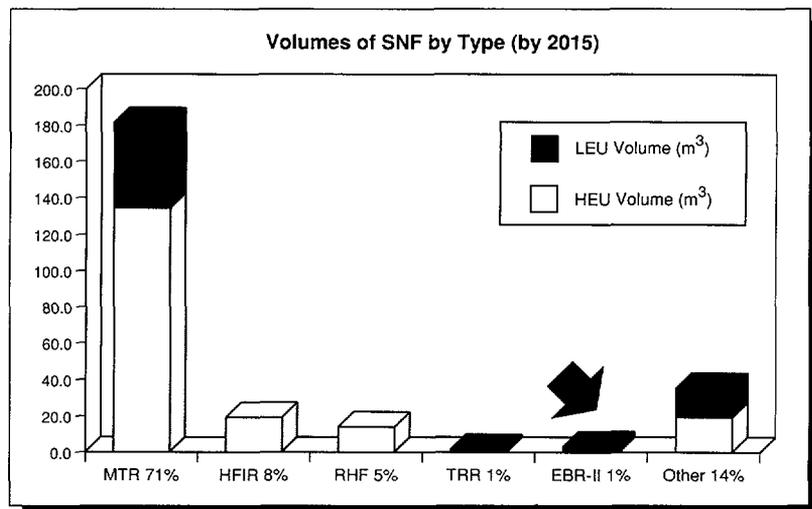
- 60 canisters currently in inventory at SRS (no additional receipts planned)

Sources and Condition

- Experimental Breeder Reactor II at Argonne National Laboratory - West
- All but one of the aluminum canisters is intact; condition of the fuel inside the canisters is unknown

Inventory

There are 60 canisters of EBR-II spent blanket fuel rods at SRS. One of these canisters is leaking and has been designated for processing in the canyon facilities at SRS. The remaining 59 EBR-II canisters comprise about 3.0 m³ (or 1% by volume) of aluminum LEU SNF.



Fuel Facts

Other Non-MTR Fuel

Fuel Description

This category includes assorted fuel elements of various configurations and sizes, not similar to MTR fuel or the other fuel types tabulated. Most is in the form of clusters or bundles of fuel pins or fuel rods of varying lengths. Several of these fuel types contain both the high-enriched and low-enriched materials. Spent fuel elements in this category make up 14% of the volume of aluminum SNF that will be managed at the Savannah River Site by 2015.

Fuels Included

	<u>No. Elements</u>	<u>Volume (m³) in 2015</u>
• Core Filter Block (LEU) (Advanced Reactivity Measurement Facility/Coupled Fast Reactivity Measurement Facility)	1	0.03
• Sandia Pulse Elements (HEU)	38	3.8
• Sterling Forest Oxide Powder (HEU)	260	4.5
• Canadian Pin Clusters (HEU and LEU)	2805	26.0
• Canadian Pin Bundles (HEU)	12	0.1
• South Korean Pin Clusters (LEU)	168	1.2

Physical Characteristics

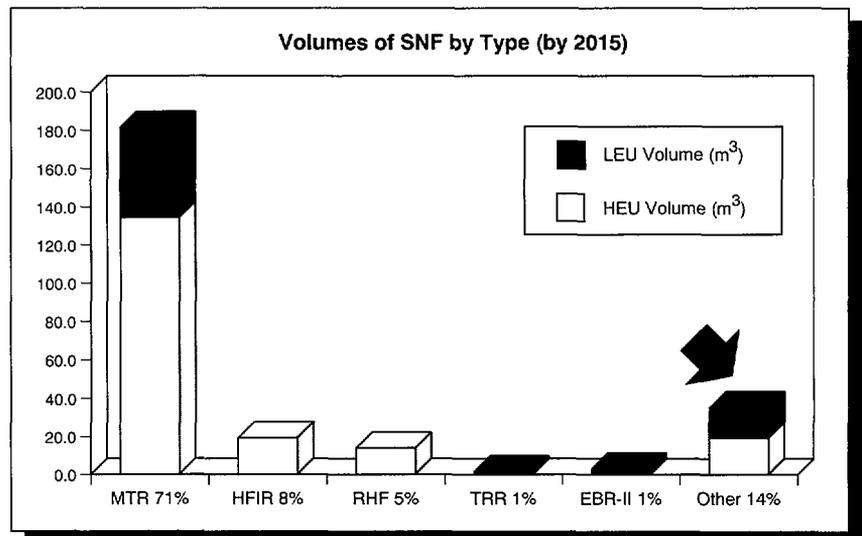
- Dimensions - Bundles and clusters of pins or rods up to 413 cm in length, core filter - 23 cm square by 76 cm long
- Materials - Uranium-aluminum or uranium oxide-aluminum; uranium metal (core filter)
- Uranium - Up to 2 kg uranium per element
- Enrichment - U-235 enrichment varies from just below 20% to as high as 93%, except the core filter which is 0.6% U-235
- Construction - Small diameter pins or rods in clusters or bundles; a portion of the Sterling Forest Oxide is powder in cans

Sources and Conditions

- Foreign research reactors in Canada and South Korea; Sandia National Laboratory, Idaho National Engineering Laboratory; and a non-DOE facility in New York.

Inventory

The only spent fuel in this category that is at the Savannah River Site currently is the Sterling Forest Oxide. By 2015, fuel in this category will amount to 19.4 m³ of the HEU SNF inventory, and 15.9 m³ of the LEU SNF inventory.



Other Non-MTR Target Material

Material Description

Special target material to be received under the Foreign Research Reactor Environmental Impact Statement. This material is configured into targets for irradiation of molybdenum to produce technetium 99 for medical purposes. The material exists as a liquid and is solidified to a powdered form, similar to the Sterling Forest Oxide fuel, prior to shipment to the U.S.

Physical Characteristics

- Dimensions - Cans for HEU similar to the Sterling Forest Oxide (6.3 cm diameter, 27.9 cm long)
- Materials - Uranium-aluminum or uranium oxide-aluminum
- Uranium - Average of 40g U-235 per can
- Enrichment - U-235 enrichment varies from just below 20% to 93%
- Form - Currently exists in liquid form, but will be solidified to a powder after removal of molybdenum and technetium

Sources and Conditions

- Foreign research reactors in Canada, Belgium, Argentina and Indonesia
- A total of 556 kg of uranium is contained in this material, 216 kg as HEU and the remainder as LEU

Inventory

- 6,750 cans will be received (6.5m³)
- None of this material is at the Savannah River Site currently
- Specific receipt rates for this material are not yet known

Fuel Facts

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3. Constructing a Path Forward - Key Building Blocks

The first step in developing a technical strategy for the treatment, packaging, and disposal of the research reactor SNF is to identify and examine, in relative isolation, the major issues. Three specific areas of investigation were considered key:

- Repository and waste form considerations
- Receipt, handling, and storage provisions
- Treatment technology alternatives

These are discussed in the following sections.

■ 3.1 Repository Considerations

■ ■ 3.1.1 Waste Form Requirements, in General

Any waste form accepted for disposal in a high-level radioactive waste geologic repository must comply with the provisions of the Nuclear Waste Policy Act (NWPA). As “spent nuclear fuel” (as defined in the NWPA), the research reactor fuel can be considered for repository disposal. Other non-fuel waste forms from the SNF treatment technologies under consideration would have to be classified by NRC as “high-level waste” in order to be considered for disposal in the repository.

Repository disposal of the waste forms from the treatment technology options will have to comply with the licensing provisions of 10 Code of Federal Regulations (CFR) Part 60 and applicable DOE Office of Civilian Radioactive Waste Management acceptance criteria. The OCRWM waste form criteria used in this study are provided in the *Waste Acceptance System Requirements Document* (for high-level waste products)² and the *Preliminary Requirements for the Disposition of DOE Spent Nuclear Fuel* (for research reactor SNF).³ These criteria are outlined in Section 4.1 of this report, along with an assessment of the degree to which the waste forms resulting from each treatment technology meet those criteria.

■ ■ 3.1.2 Criticality Control

As noted in Section 2, much of the research reactor fuel is HEU. For that reason, criticality control is an important consideration in establishing waste form suitability for this material. It has been shown that the probability of occurrence of a criticality event within a repository is quite small and that the dose consequences from such an occurrence would be negligible.⁴ However, these results are not presently reflected in regulatory requirements. Therefore, for

purposes of this study, the Team based its comparison of alternatives on the premise that the amount of fissile material that could be placed in any disposal canister should be limited to that which would preclude a repository criticality event, with high likelihood. This approach is consistent with that currently taken for the geologic disposal of commercial SNF.

Presently, NRC regulations for criticality control state that “the calculated effective multiplication factor (k_{eff}) must be sufficiently below unity to show at least a 5% margin, after allowance for the bias in the method of calculation and the uncertainty in the experiments used to validate the method of calculation.”⁵ OCRWM has recommended that NRC revise this requirement to allow for criticality control to be demonstrated on a probabilistic risk analysis approach.^{6,7} To achieve true cost-effective risk control in the repository post-closure, the criticality aspects of 10 CFR 60 must be changed.

With this in mind, the Team established a range of planning values for fissile mass loadings within a waste package, based on package configuration and the enrichment level of the contained SNF. Three enrichment categories are addressed: HEU ($\geq 20\%$ U-235), LEU (between 2% and 20% U-235), and V(ery) LEU ($\leq 2\%$ U-235). For these three cases, the regulatory k_{eff} limit of 0.93 (i.e., 0.95, less allowance for uncertainty and bias) translates to a maximum per-package fissile content as follows:^h

- For HEU, each package is limited to 14.4 kg U-235.
- For LEU, each package is limited to 43 kg U-235.
- For VLEU, each package is limited to 200 kg U-235.

This approach is considered to be conservative, based on current models and information, and is similar in concept to that expected to be applied to commercial SNF.^{i,8} As more complete technical information becomes available, such as waste form characteristics, waste package detailed design, and repository environment, it may be possible to increase the permissible per-package fissile mass loading.

This fissile mass loading analysis and further comment on the criticality aspects of 10 CFR 60 are included in detail in Appendix B of this report.

^h In these calculations, no burnup credit has been applied. For research reactor SNF (unlike commercial fuel) burnup credit does not yield a great benefit in criticality analyses because of initial high enrichment and relatively low average atom percent burnup. (Typically, less than 20% of the initial U-235 atoms are fissioned, yielding end enrichments in the 60-70% range.) Furthermore, because of the variety of research reactor fuel types and the variation in their operating histories, obtaining burnup credit records for research reactor fuels may be impractical.

ⁱ The analysis takes into account the anticipated degradation of aluminum. The calculated per-package fissile loadings identified in this study for aluminum-based SNF, therefore, may not apply to more robust waste forms such as commercial SNF or naval reactor SNF.

3.2 Provisions for SNF Receipt, Handling, and Storage

All of the treatment technologies will require a complete, integrated site infrastructure with capability for SNF receipt, handling, characterization, interim storage, and preparation for disposal. In many respects these capabilities would be similar for all of the treatment technologies and all of the fuel types, but in some cases the requirements differ. Figure 3.2.-1 is a simplified process flow model showing the overall site infrastructure required. The following briefly describes these functions and the facilities needed for their performance.

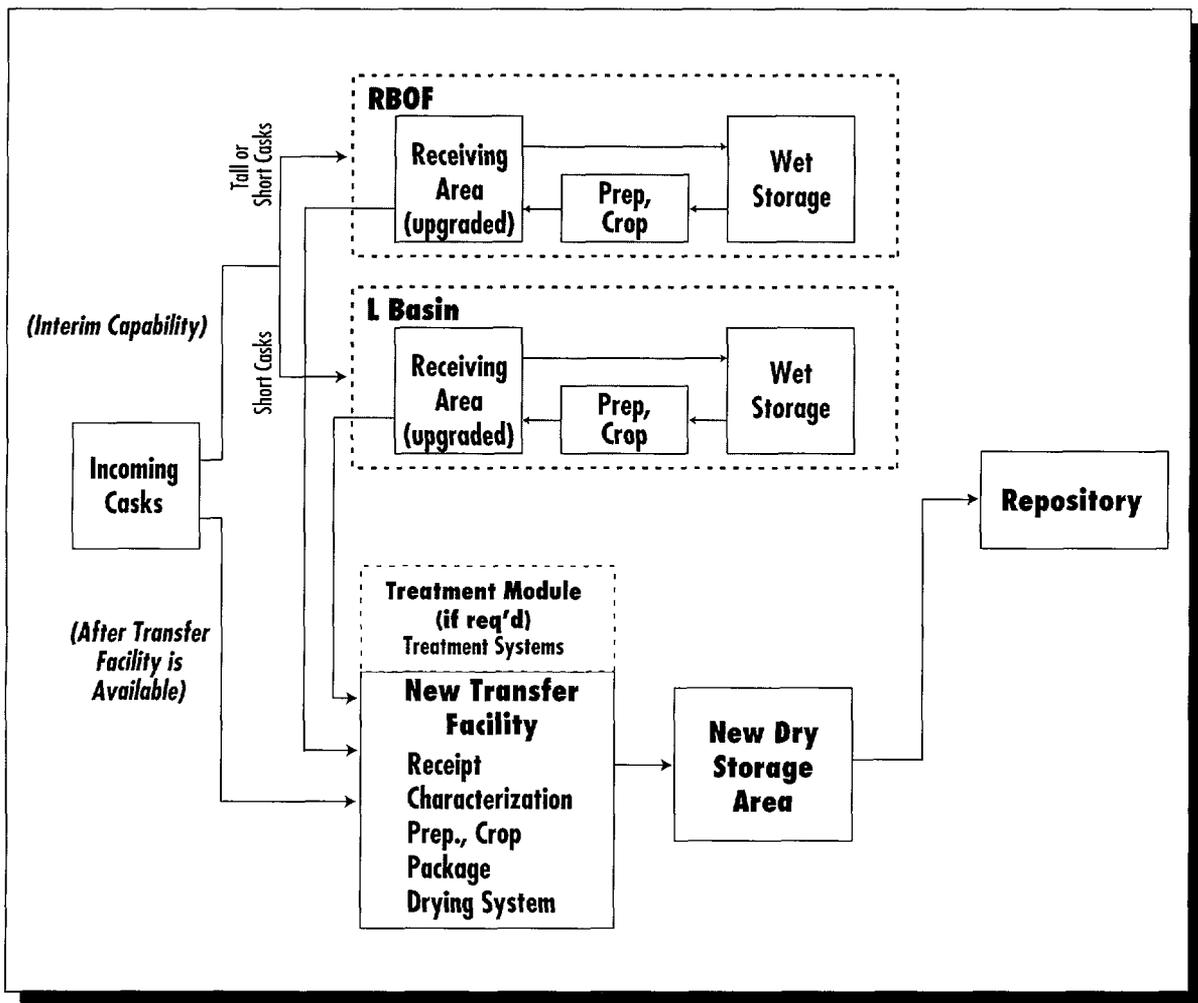


Figure 3.2-1 SNF Receipt, Handling, and Storage

Receipt of SNF Shipments

It was assumed that all SNF will be transported in casks similar to the existing fleet. These will be received at existing wet basin facilities until new transfer and handling facilities are operational.

Wet Basin Storage

Existing wet basins at SRS, specifically the Receiving Basin for Offsite Fuels (RBOF) and the L-Reactor Basin (L-Basin), must be used for receipt and interim storage, at least until new facilities can be constructed. Expected receipts during 1996 will bring RBOF to its full capacity. L-Basin will be used for all new interim storage beginning in 1997.

Until new storage capacity is brought on-line, the inventory of SNF assemblies in L-Basin will continue to grow. Once a new transfer facility (discussed below) becomes available, the L-Basin and RBOF would be unloaded as quickly as the new process would permit, eliminating dependence at SRS on wet storage for SNF. Early availability of the transfer facility permits removal (from RBOF), drying, and packaging of existing stainless steel and zirconium-clad SNF. This SNF has been identified for shipment to the INEL under the Programmatic EIS. Following this effort, transfer facility operations would focus on aluminum-based SNF in conjunction with startup of the treatment module.

Characterization

For the purposes of this study, it was assumed that only nominal characterization activities would be required upon receipt. This implies that the SNF would arrive with adequate documentation and history of its use. The only activities assumed upon receipt are sipping each cask (a process used to test for leaking fuel elements), a visual inspection of each SNF assembly, and a gamma scan of each assembly. Any additional characterization would be limited to analysis of a few assemblies for each fuel type for isotopic content and leaching characteristics, followed by nondestructive examination to estimate performance for each fuel type. If additional characterization were required, some of the treatment technologies would be better able to accommodate such activities than others.

SNF Transfer Operations

Facilities for new treatment technologies should include the capability to receive shipments from off-site or from the existing wet storage basins, and be sized to receive projected off-site shipments and to de-inventory the backlog of basin storage after the treatment process comes on-line. These functions would be centralized in a new transfer facility, designed to handle all incoming and outgoing material, and to interface with the selected treatment technology.

The functions of the new transfer facility, including receipt and unloading of shipping casks, transfer of treatment or conditioning products to packages, sealing of those packages, and transfer to interim storage, are common to all options investigated. Integral to the transfer facility would be some lag storage (dry) that would be used to facilitate the packaging, characterization, and treatment steps. Situated adjacent to the transfer facility would be additional dry storage modules that could be added on a phased schedule. Figures 3.2-2 and 3.2-3 show a conceptual configuration for the transfer facility.

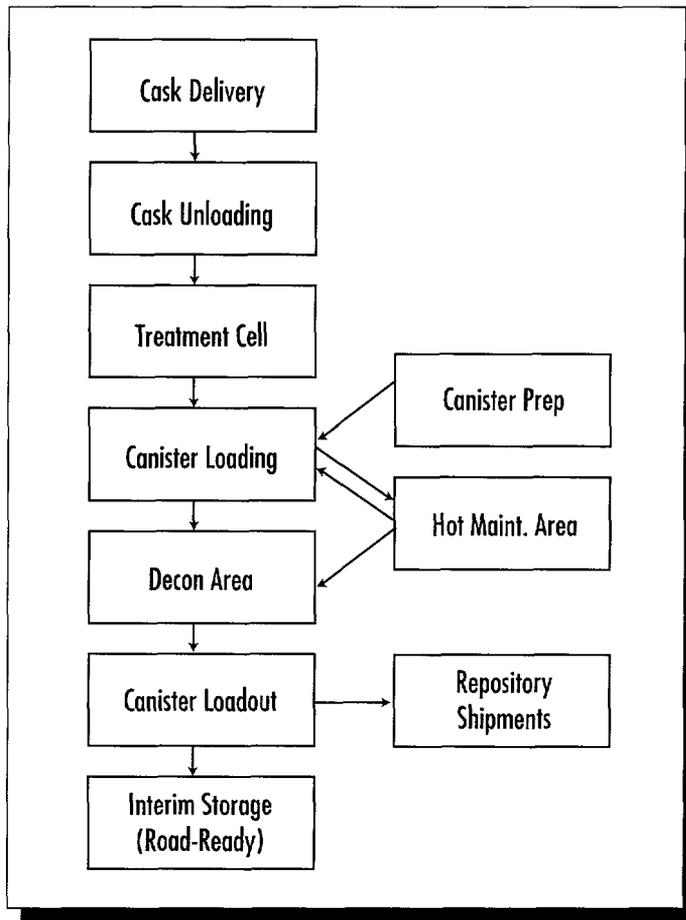


Figure 3.2-2 Transfer Facility Functions

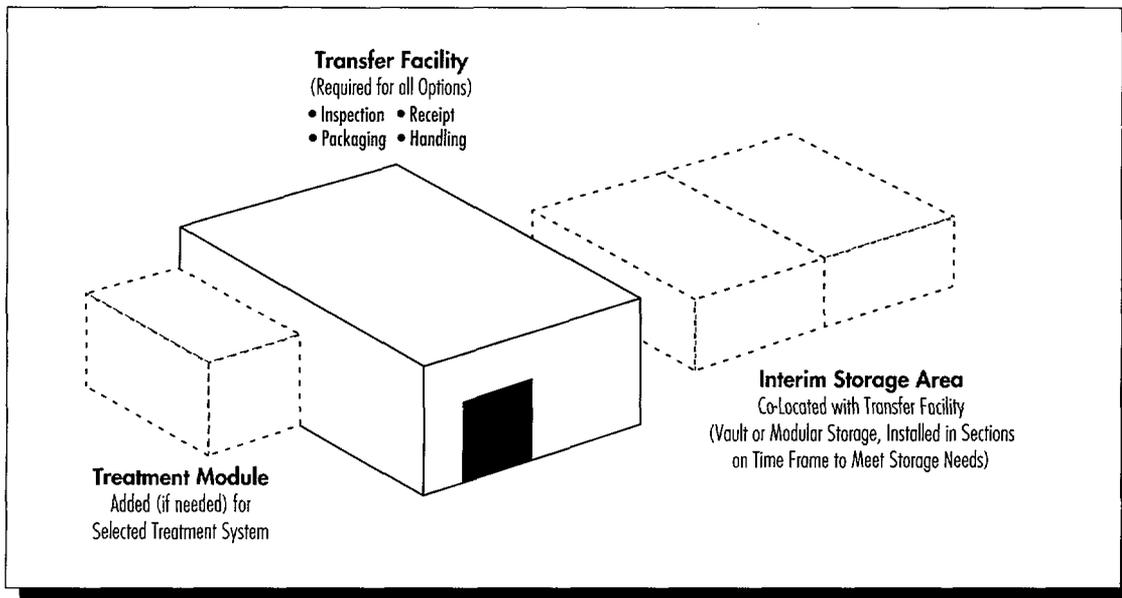


Figure 3.2-3 Transfer Facility Concept

The Team also evaluated the feasibility of conducting all operations from existing wet basin facilities instead of constructing a transfer facility, particularly for the direct disposal options for which a new treatment facility would not be required. Based on historical handling durations for cask receipts and estimated duration for loading, sealing and transfer of dry storage packages, there does not appear to be sufficient throughput capability available in RBOF and L-Basin to handle all required receipts and transfers. As a result, L-Basin would have to remain in operation past 2021, and RBOF would remain in operation until 2035. The high operating costs of extended wet basin operations, contrasted to the significantly lower capital and operating costs for a new facility, argue for early availability of a new dry transfer capability.

Since dry transfer operations are common to all new treatment technologies, engineering and construction of a transfer facility can proceed in the near term, and prior to treatment technology selection(s). This approach would provide for phased modular implementation, including addition of a treatment module when a treatment technology is selected. This would allow early start of operations for the direct disposal concepts, and timely implementation of the treatment technologies selected for all or part of the SNF, along with cost savings and technical merit of early closure of the wet basins. This approach has not been evaluated in detail in this study, but deserves further evaluation. Additional information on variations to the transfer facility is provided in Appendix C of this report.

Interim Dry Storage

Each treatment technology will require interim storage of its product in a road-ready condition until the repository is available. Although all commercially available dry storage concepts are likely to be feasible and practical for this SNF, the conceptual evaluations in this study suggest that the modular dry vault storage concept would be very efficient; this has been used as the baseline dry storage concept for this study. The selection of dry storage technology would likely be made via competitive bidding, based on the actual design parameters for the SNF packages produced by the chosen technology.

■ 3.3 Candidate Treatment and Disposal Technologies

The treatment and disposal technologies evaluated by the Team included those identified in the DOE SNF Technology Integration Plan⁹ along with several technology concepts recently identified within DOE. The following sections briefly describe the technologies evaluated. These are categorized according to waste form and treatment complexity. Along with these general descriptions, a “treatment technology synopsis” sheet summarizes information for each candidate relevant to the comparative analysis, e.g., waste form, technical maturity, package configuration, handling considerations, etc. (see pages 35-43).

■ ■ 3.3.1 Direct Disposal Technologies

Direct disposal is the most technically simple approach considered. The SNF would be dried and placed in a canister for interim storage, then transported to the repository and placed in a waste package for disposal. Two concepts were evaluated, direct disposal in small packages and direct disposal in co-disposal packages.

SNF may require characterization to meet the storage, transportation, and repository waste acceptance requirements. With the exception of cropping the ends of the fuel assemblies, no physical or chemical changes would be made to the SNF. The SNF would be dried sufficiently to limit corrosion during storage, and to preclude any excessive gas generation. A new facility would be needed for the drying and packaging of the SNF.

The excess SNF structural material would be removed to conserve space. The containers would be sealed prior to being placed into interim storage, awaiting shipment to the repository. During interim storage some buildup of pressure is possible and canisters may need to be returned to the transfer facility for venting prior to repository disposal. Direct disposal does not involve separation of the fission products from the uranium, so the fuel, as packaged for disposal, is self-protecting and would not present proliferation concerns.

Direct Disposal, in Small Packages

For this direct disposal option, the SNF is placed in small packages for individual burial in the repository. The U-235 mass planning values restrict the quantity of SNF that may be placed in a canister. The small packages are placed in the repository between waste packages of commercial SNF (Figure 3.3-1).

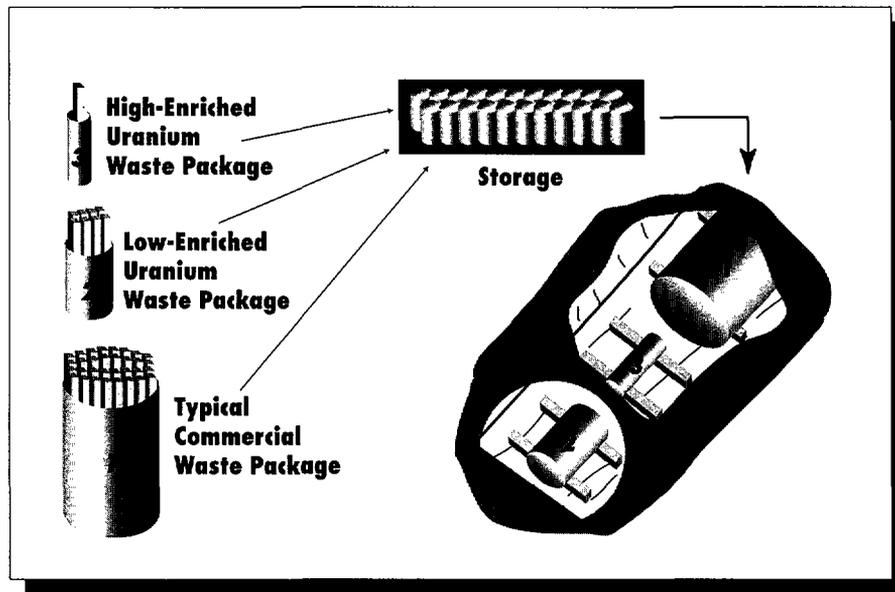


Figure 3.3-1 Direct Disposal, in Small Packages

Direct Disposal in Co-Disposal Packages

For this option, the SNF would be placed in small canisters that would be disposed by placement in repository waste packages which contain HLW glass logs. Fuel quantities may be limited to satisfy repository criticality requirements. As currently envisioned, the repository waste package design would permit five 24-inch diameter glass logs to be packaged in a circular configuration, leaving a center cavity for an SNF canister approximately 17 inches in diameter. The SNF canisters would be placed in the repository waste packages at the repository. The co-disposal concept would permit repository disposal of all research reactor fuel, with no additional repository space beyond that required for disposal of HLW glass logs themselves (Figure 3.3-2).

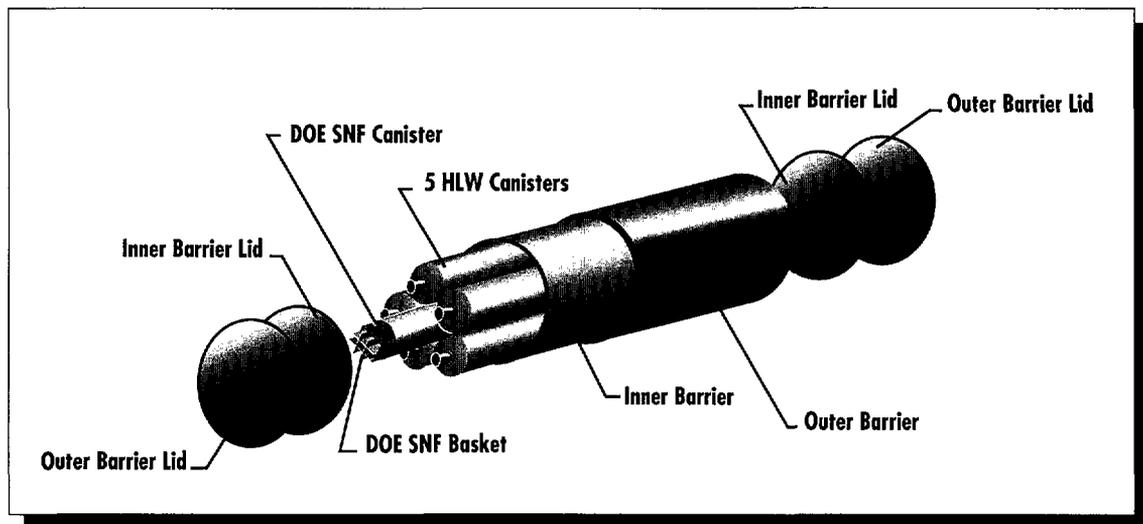


Figure 3.3-2 Co-Disposal Waste Package Concept

■ ■ 3.3.2 HEU Dilution Technologies

Major issues associated with the direct disposal of research reactor fuel in a repository include nonproliferation and criticality control, both of which may be of concern for HEU. At this point, consideration must be given to the potential desirability and/or regulatory necessity of diluting HEU fuel to below 20% enrichment. As noted in Section 3.1.2, the probability of a criticality, already low for SNF stored as HEU, could be lowered further by reducing enrichment. The Team evaluated HEU dilution methods for purposes of reducing the effective enrichment of the waste packages, thereby improving package efficiency and rendering the packages less attractive for diversion.

The most practical way to reduce effective enrichment is to mix the fuel with natural or depleted uranium. Natural uranium has 0.7% U-235, and depleted uranium typically has 0.4% or less U-235. The remainder of the uranium in both cases is primarily U-238, which is not readily fissionable and cannot support a nuclear criticality. Applying the fissile material limits described in Section 3.1.2, dilution to less than 20% (U-235) would permit 43 kg of U-235 to be placed in each canister, and dilution to less than 2% would permit 200 kg of U-235 to be placed in each canister. The addition of the diluents increases the volume of the material to be disposed, but the increase in the allowable amount of SNF in each canister results in a potential decrease in the total number of canisters required.

Two methods of diluting the U-235 were evaluated: sandwiching pressed depleted uranium in layers with the SNF, and melting the SNF in a mixture with depleted uranium. The spacing (gaps) between the fuel plates or rods (used for coolant flow during reactor operations) represents much of the SNF volume. By mechanically pressing the fuel or by melting the fuel, the gaps can be eliminated and the volume of the fuel reduced. In either case the SNF has to be adequately characterized to meet the repository requirements. In the case of melting, it may be possible to characterize the melted solution instead of the fuel.

No separation of the fission products from the uranium occurs in the dilution process so the fuel, as packaged for disposal, remains self-protecting.

Press and Dilute/Poison

In this option, the SNF assemblies would be flattened in a mechanical press to eliminate the air gaps. Depleted uranium would then be pressed around the flattened fuel to reduce the effective enrichment of the composite to either 20% or 2%, as desired. Several flattened assemblies could be placed into one unit, making a "sandwich." The sandwich would be placed into an interim storage canister and later shipped to the repository for disposal. Proper dimensioning of the interim storage canister would permit its use in a co-disposal waste package as described above.

Melt and Dilute

The SNF would be melted in a crucible with depleted uranium. This would reduce the effective enrichment to 20% (or further to 2%, similar to the Press and Dilute option). Because uranium oxides and silicides do not form alloys with the uranium metal, the melted solution would be

mixed to achieve homogeneity. The melting would release volatile and semi-volatile fission products that would have to be captured and disposed. As with the Press and Dilute option, proper dimensioning of the interim storage canister would also permit its use in a co-disposal waste package.

■ ■ 3.3.3 Advanced Treatment Technologies

The advanced treatment technologies evaluated by the Team would produce more resilient waste forms and fewer waste packages than the direct disposal and dilution options. These methods have been in development for potential application to other SNF, and are evaluated here for applicability to aluminum-based SNF. Each technology was identified in the DOE SNF Technology Integration Plan.

The technologies in this category reduce the enrichment to less than 20% or remove the uranium, and produce either a glass or ceramic waste form. This decreases the risk of the SNF not being acceptable for repository disposal. In each technology the SNF matrix is changed (i.e., by melting and oxidation, by acid dissolution, etc.) followed by the production of a glass or ceramic waste. This change in the physical makeup of the SNF significantly reduces the need for pre-treatment characterization. It is expected that the resultant waste form can be characterized by either process or batch sampling.

The U-235 dilution eliminates proliferation concern.

All of the new technologies would require the construction of a facility to house the operation.

Plasma Arc Treatment (Vitreous Ceramic)

This process uses the high temperatures generated by a plasma arc to oxidize the metal components of the SNF. A rotating furnace mixes the molten materials into a homogeneous mixture. Depleted uranium is added to the furnace to reduce the U-235 enrichment in the waste form. A minimal quantity of other ceramic forming materials needs to be added. The melt is poured into a container, forming a ceramic solid. The ceramic has properties that are as good as or better than borosilicate glass with respect to release of radioactive materials. Only minimal characterization of the SNF is required prior to treatment.

The process is versatile in that other waste materials requiring disposal can be added in-process, and captured in the ceramic waste form.

The process has been used successfully to treat non-radioactive and low-level radioactive waste at several locations in the United States. Development is needed to apply the process to SNF because of the much higher fission product content. The offgas that would be generated needs to be collected and recycled. The ceramic waste form can accommodate higher uranium content than borosilicate glass without sacrificing performance. No work on qualifying the waste form has been performed.

GMODS (Borosilicate Glass)

The Glass Material Oxidation and Dissolution System (GMODS) technology converts SNF directly to a uranium-containing borosilicate glass. The process uses lead dioxide to oxidize the metals in the SNF so that they are soluble in the glass. (The lead metal is later oxidized in a separate process and recycled to the glass melter and, therefore, is not part of the waste form.) Glass materials are added to achieve the desired glass properties. Depleted uranium is added to reduce the U-235 enrichment in the waste form. The glass can be poured directly into glass logs, or can be formed into glass marbles, which can be mixed with other glass types at other locations. Only minimal characterization of the SNF is required prior to treatment.

The process has been successfully demonstrated on simulated SNF materials in a laboratory. Further development is needed to apply the process to actual SNF. This process could be used on all types of SNF, including graphite, with no changes to the system. Although no work has been done to qualify the uranium glass for disposal in the repository, that material is very similar to DWPF borosilicate glass, and its qualification is not expected to be difficult.

Dissolve and Vitrify (Borosilicate Glass)

This technology is similar to that of conventional processing in that it involves dissolving the fuel in acid. Unlike conventional processing, however, the fissile materials are not separated from the fission products. Depleted uranium is added to reduce the U-235 enrichment in the waste form. The dissolved mixture is then fed directly to a vitrification plant for conversion to a uranium containing borosilicate glass.

The dissolution process has been demonstrated successfully for many years. The production of glass in DWPF has just begun. Development is needed to apply the process to uranium solutions. No work on qualifying the uranium glass for the repository has been performed, but the glass is expected to be similar to DWPF borosilicate glass.

Electrometallurgical Treatment

In the Electrometallurgical Treatment option, the SNF is shredded and added to a melter, where it is converted to a metal ingot. The ingot is placed in an electrorefiner, where the aluminum is removed as a low-level waste stream. The material is next placed into another electrorefiner where the uranium is removed. The remainder of the material is oxidized in a furnace and dissolved into glass, producing a borosilicate glass similar to the DWPF glass. As part of the process, the uranium is blended down to LEU to resolve proliferation concerns. Also, because SNF is converted to a new form, requirements for pre-treatment characterization of the SNF will be minimal.

Electrometallurgical Treatment is the only alternative considered in this evaluation that involves separation of uranium and HLW. Removal of the fissile uranium eliminates any concerns about a repository nuclear criticality.^j

The uranium removal sub-process has been well tested, both hot and cold, and a test on actual SNF is scheduled to begin in Idaho in late 1996. Development of the newly proposed electrorefining steps for aluminum will need further analysis and testing. Previous reviews of the electrometallurgical process, such as one completed by the National Research Council,¹⁰ report that aluminum fuel characteristics would significantly challenge containment and effluent treatment for the electrorefining process. However, the proposed process for treating the aluminum fuels has been significantly revised since those reviews were completed, and the aluminum characteristics may not be as challenging to the electrometallurgical process as previously assumed.^k

■ ■ 3.3.4 Other Concepts

Initially, eleven alternative technologies were proposed as treatment candidates. The Team's initial screening determined that three of the proposed technologies did not pass the "must" requirements (see Table 4.4-1) and did not warrant further evaluation for this SNF.

Chloride Volatility

This concept has been the subject of several paper studies but no experimental work has been completed. In concept, chlorine gas would react with the SNF at high temperatures, converting all of the materials to volatile chlorides. The gasses would be separated from one another by scrubbing and fractional distillation. The method allows a complete separation of the SNF elements. Because of the time and money that would be required to develop the method, it was not evaluated further.

Chop and Dilute/Poison

For this concept the SNF would be mechanically shredded and mixed with depleted uranium. Because this method is similar to the Press and Dilute and the Melt and Dilute technologies, both of which were deemed superior in their ability to control the treatment steps needed to produce a diluted waste form, the Chop and Dilute technology was not evaluated further.

^j Uranium content in SNF influences disposal volume in two ways. First, the uranium can be a substantial part of the metallic volume of the SNF (as much as 90% for uranium metal fuels). Second, many of the proposed waste forms are limited by the concentration of uranium that they can contain. For example, borosilicate glass can contain only 10–20% uranium. Therefore, removal of the uranium and aluminum leaves HLW material (fission products and actinides) which requires far less disposal volume than uranium- and aluminum-bearing wastes.

^k The National Research Council reviewed a single step electrorefining process of uranium that was complicated by the presence of aluminum. The revised concept that was reviewed by the Team calls for separating the aluminum prior to removal of the uranium. The revised concept was the result of work being conducted for single-pass reactor fuel at Hanford and proposed stabilization techniques being evaluated for the Molten Salt Reactor Experiment at Oak Ridge.

Can-in-Canister

This technology is based on a similar application being evaluated for plutonium disposal and involves the placement of intact SNF elements into cans. These cans would then be placed in canisters into which HLW glass is poured to form solidified units. Applying this technology to aluminum-based SNF raises several technical problems related to the low melting point of the aluminum. The Direct Disposal (co-disposal packages) concept was considered by the Team to be a better approach, so the Can-in-Canister concept was not evaluated further.

■ ■ 3.3.5 Variations Considered

Several variations on the above concepts for packaging, handling, and treatment were identified by the Team. These were not reviewed in detail and may warrant further evaluation.

Point-of-Origin Packaging

Small casks, designed for interim dry storage and transportation, would be loaded at the shipping reactor site, then transported to SRS for storage on a pad awaiting transport to the repository. Receipt and handling activities at SRS would be minimized.

Use of DWPF Glass Waste Storage Building (GWSB)

Extra GWSB storage space does not appear to be available; thus, a savings is not apparent. The cost of building additional GWSB storage should be approximately the same as building the interim storage facility. Also, the GWSB is not designed for storage of fissile materials.

Retain Aluminum-Based SNF at INEL

Not receiving the INEL SNF potentially reduces the demand placed on the packaging, handling, and treatment facilities at SRS. This raises larger issues associated with intersite SNF transfers.

■ 3.4 Comparative Scenario: Processing/Co-Disposal

In order to produce a thorough evaluation, the team felt it advisable to compare the alternative treatment technologies to a reference technology having well-known evaluation parameters. The Team constructed a hybrid scenario that included near-term chemical processing in the existing SRS canyons through 2008, followed by direct co-disposal of the remaining SNF.

The Plutonium Uranium Recovery Extraction (PUREX) fuel processing technology is the reference technology against which the candidate technologies are compared. The PUREX technology involves the acid dissolution of the aluminum-based SNF with the uranium separated from the fission products. The uranium is recovered for reuse, and mixed with depleted uranium to reduce the U-235 enrichment to less than 20%. The fission products, with small quantities of plutonium, are sent to the HLW tanks where they become feed to the DWPF for conversion to glass logs.

Through approximately 2002, SRS processing capability will be applied to material stabilization and processing of the Mk-16 and Mk-22 fuel and Mk-31 targets, and processing of a limited amount of existing research reactor SNF deemed to be a potential safety concern. Costs for processing additional aluminum-based SNF during this time frame would be incremental. Following this processing campaign, additional SNF could be processed, but costs would increase as the base mission is completed. The comparative scenario assumed that processing capability would be available until 2008, at which time there would be no inventory backlog. Any SNF received after 2008 is assumed to be managed by the Direct Co-Disposal technology.

Treatment Technology Synopsis

Direct Disposal

Conceptual Description

The SNF is cropped and repackaged into small canisters, sized and loaded to accommodate U-235 mass limits. Canisters are of three lengths (5', 10' and 16'), all 2' diameter, fabricated of steel with neutron poison inserts as required. The SNF is vacuum dried to remove free water, but otherwise unchanged from its current condition. The canisters are dry-stored until the repository is available, vented if necessary, then transported and placed in repository overpacks for emplacement.

Products

Approximately 1100 packages configured as described above.

Secondary Streams

None

Criticality Implications

HEU is stored in individual packages; U-235 mass content limits, spatial distribution, and poisons (if used) reduce the probability of a criticality in the repository.

Proliferation Resistance

The storage and disposal package contains HEU with no fission product removal.

Technical Maturity

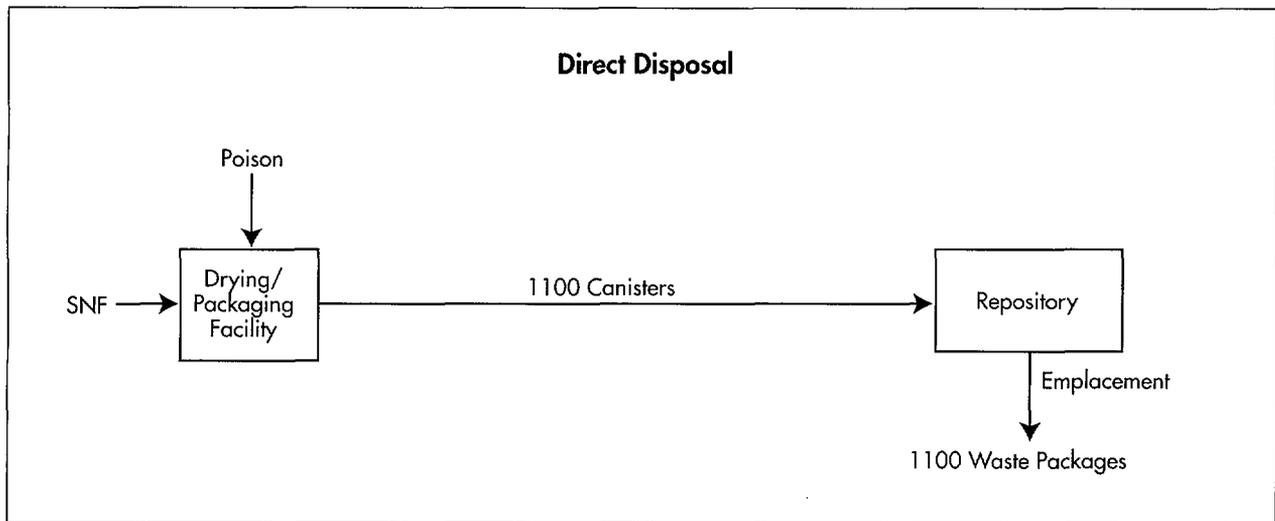
This method employs simple, proven technology. No long-term equipment development is required. Analytical work is needed to confirm waste form suitability, particularly aluminum storage and HEU repository issues. Production availability by 2000.

Cost

See cost summary sheet. The total is driven by disposal costs of a large number of individually stored packages.

Other

There may be no simple way to sample and characterize the fuel.
 Method may not be suitable for powdered fuels (Sterling Forest Oxide & molybdenum targets).
 There is some uncertainty about licensability of direct disposal of HEU and metallic forms.
 A variation would allow the fuel to be characterized, prepackaged, and dried at the reactor sites prior to shipment to SRS for interim storage.



Treatment Technology Synopsis

Direct Co-Disposal

Conceptual Description

The SNF is cropped and placed into small canisters, sized to fit in the center space of a repository waste package which contains HLW glass logs. The co-disposal canisters are nominally 17 in. wide, 10 ft. long, and loaded to accommodate U-235 mass limits. The canisters are fabricated of steel with neutron poison inserts as required. The SNF is vacuum dried to remove free water, but otherwise unchanged from its current condition. The packages are stored until the repository is available, then transported and co-loaded with HLW glass logs in repository overpacks for emplacement.

Products

Approximately 1400 canisters, configured as described above.

Secondary Streams

None

Criticality Implications

HEU is stored in individual packages; U-235 mass content limits, spatial distribution, and poisons (if used) reduce the probability of a criticality in the repository. The HLW glass logs within the repository waste package may have value in the criticality analysis.

Proliferation Resistance

The storage and disposal package contains HEU with no fission product removal.

Technical Maturity

This method employs simple, proven technology. No long term equipment development is required. Analytical work is needed to confirm waste form suitability, particularly aluminum storage and HEU repository issues. Production availability by 2000.

Cost

See cost summary sheet. Significant benefit in cost sharing repository package.

Other

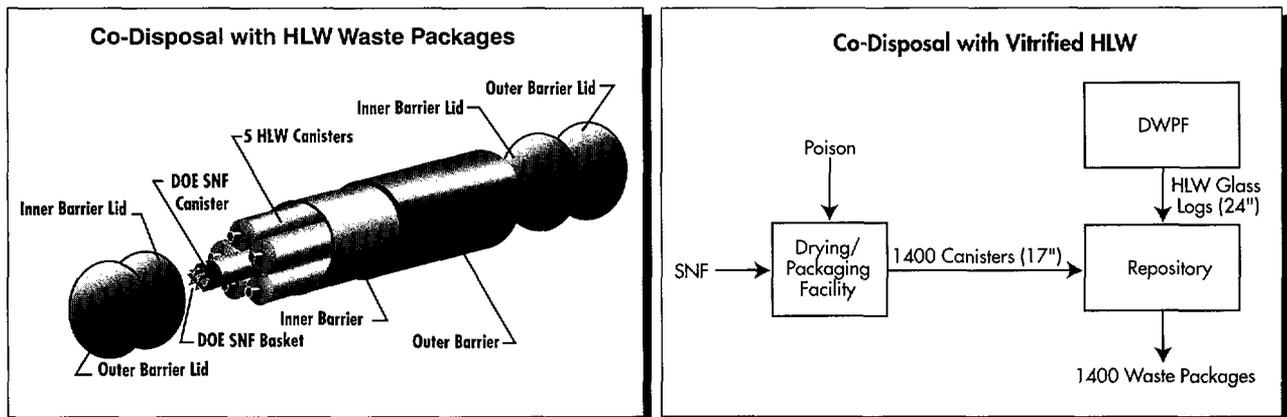
There may be no simple way to sample and characterize the fuel.

Method may not be suitable for powdered fuels (Sterling Forest Oxide & molybdenum targets).

There is some uncertainty about licensability of direct disposal of HEU and metallic forms.

Some fuels have dimensions requiring different geometry containers.

A variation would allow the fuel to be characterized, prepackaged and dried at the reactor sites prior to shipment.



Treatment Technology Synopsis

Press and Dilute/Poison

Conceptual Description

The fuel is cropped and cold-vacuum dried. The assemblies are pressed flat and sandwiched between layers consisting of either depleted uranium-aluminum alloy, depleted uranium (DU), or depleted uranium oxide (form to be selected after further study), to produce dimensionally uniform packages with a composite enrichment of 2% or 20% U-235 (enrichment to be selected after further study).

Products

Approximately 400 (20% U-235) cylindrical steel canisters (17 in. by 10 ft.) containing heterogeneous layers of pressed fuel and diluents, suitable for co-disposal packaging. As an alternative, approximately 140 large packages (dimensions similar to multi-purpose canister [MPC]) could be produced, suitable for direct disposal.

Secondary Streams

None

Criticality Implications

2% and/or 20% effective enrichment packages minimize criticality concerns. A potential long term criticality issue may be associated with non-uniform degradation. For 20% enrichment, poison may be required.

Proliferation Resistance

The average enrichment is reduced, but the HEU and DU are joined mechanically (not chemically bonded). The fission products will not be separated from the uranium.

Technical Maturity

The process is mechanical and relatively simple. The waste form (heterogeneous) will need to be qualified for disposal. Production availability by 2003.

Cost

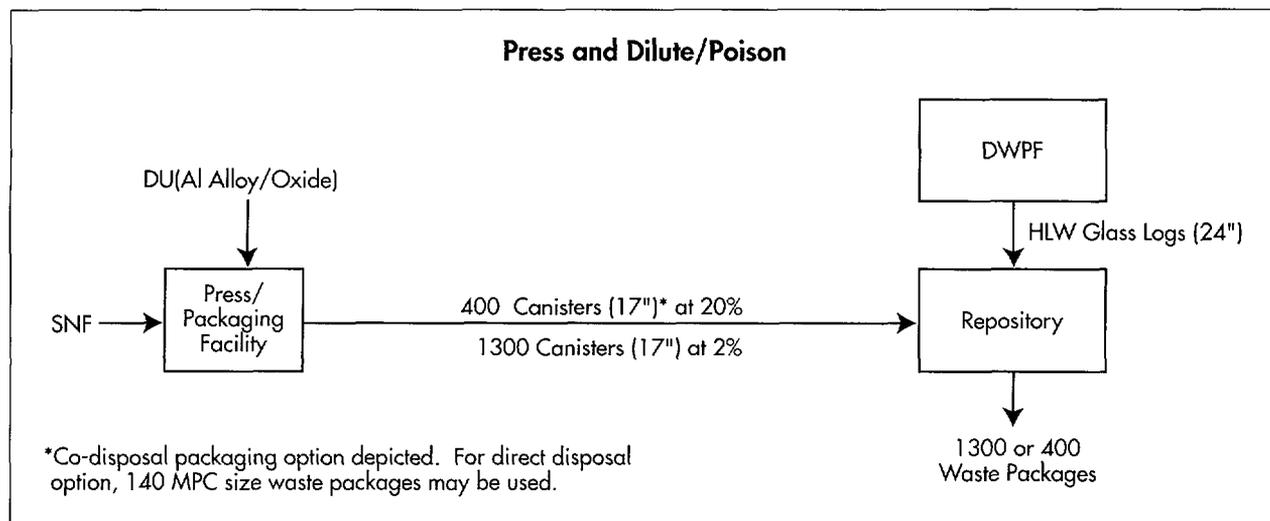
See cost summary sheet.

Other

There may be no simple way to sample and characterize the fuel.

Method may not be suitable for powdered fuels (Sterling Forest Oxide & molybdenum targets).

There is some uncertainty about licensability of direct disposal of HEU and metallic forms.



Treatment Technology Synopsis

Melt and Dilute

Conceptual Description

The fuel is cropped and melted with depleted uranium (DU) in a crucible, to produce a mixture with uranium content less than 20% U-235. The melted material is solidified in the crucible and placed along with the crucible in a cylindrical steel canister suitable for co-disposal with HLW glass logs.

Products

Approximately 400 canisters, configured as described above.

Secondary Streams

The melting process produces an offgas of volatile and semi-volatile fission products (e.g. cesium) that must be captured and disposed

Criticality Implications

The quantity being melted must be controlled to reduce the potential for in-process criticality. The lower enrichment reduces criticality concerns in repository.

Proliferation Resistance

The storage and disposal package contains LEU with no fission product removal.

Technical Maturity

The process is relatively simple and well understood. Production availability by 2003.

Cost

See cost summary sheet.

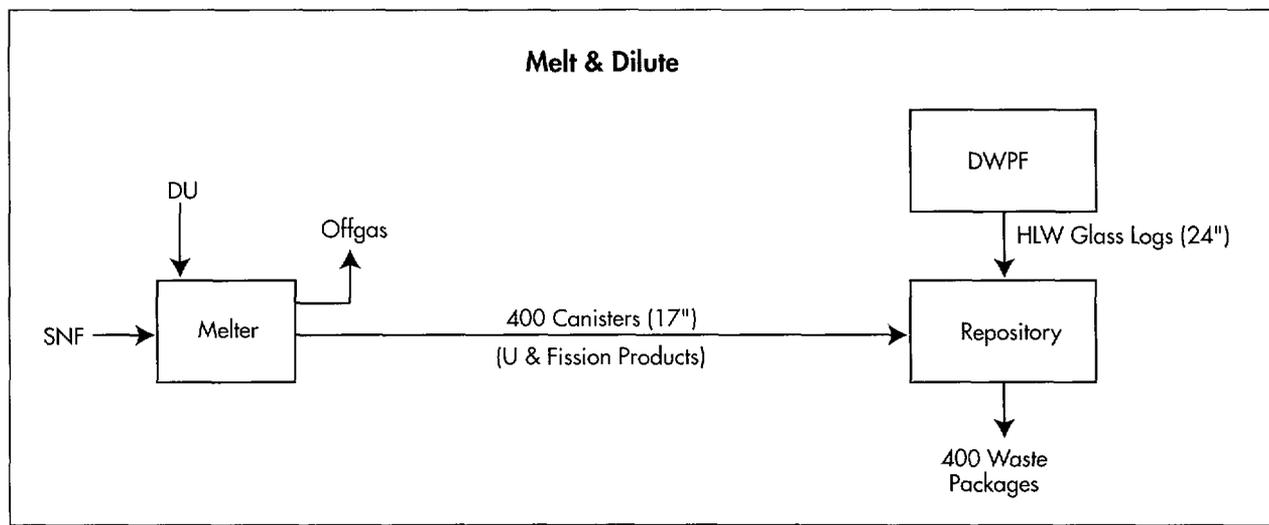
Other

There may be no simple way to sample and characterize the fuel

The oxides will not dissolve in melter (also silicides may not) and therefore technical development is required. An alternate treatment method may be necessary for oxides and silicides.

Must dispose of crucible.

The waste form needs to be qualified.



Treatment Technology Synopsis

Plasma Arc Treatment

Conceptual Description

A batch process using a centrifugal furnace, in which the fuel is melted and oxidized with depleted uranium (DU), to produce a vitreous ceramic with a U-235 enrichment of 20%. (Borosilicate glass can also be produced.)

Products

Approximately 400 steel canisters (determined by U-235 loading), sized for co-disposal.

Secondary Streams

The melting process produces an offgas of volatile and semi-volatile fission products (e.g. cesium) that must be captured and disposed.

Criticality Implications

The quantity being melted must be controlled to reduce the potential for in-process criticality. The lower enrichment and waste form stability reduce criticality concerns in repository.

Proliferation Resistance

The storage and disposal package contains LEU with no fission products removal. The ceramic form makes uranium recovery difficult.

Technical Maturity

Demonstrated in part, in non-fissile, low fission product loading application. Substantial (two decades) non-radioactive, commercial plasma experience. Development required. Production availability in 2006 time frame.

Cost

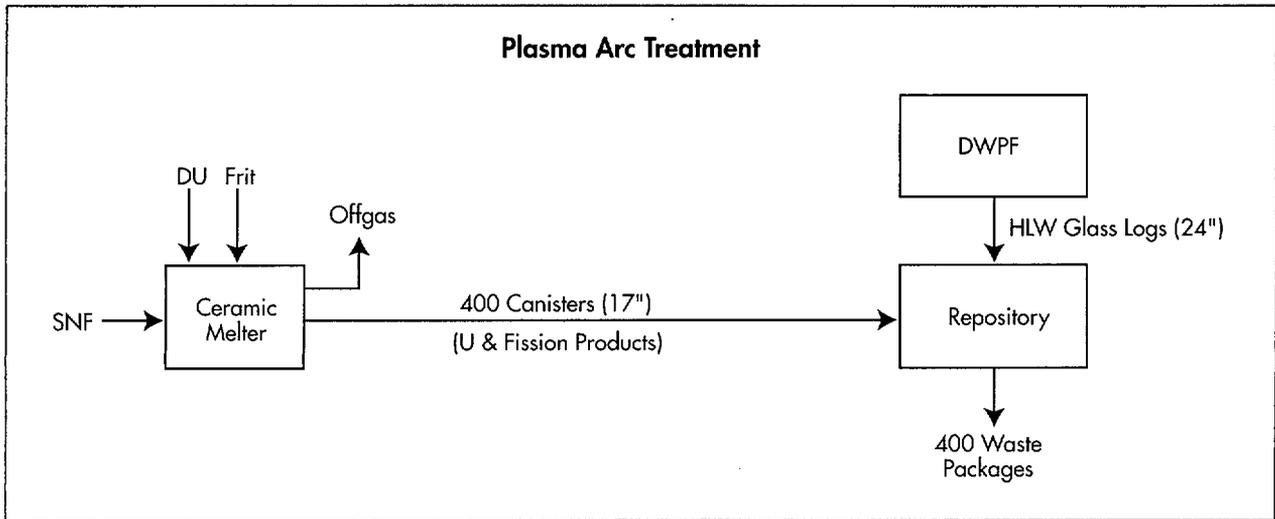
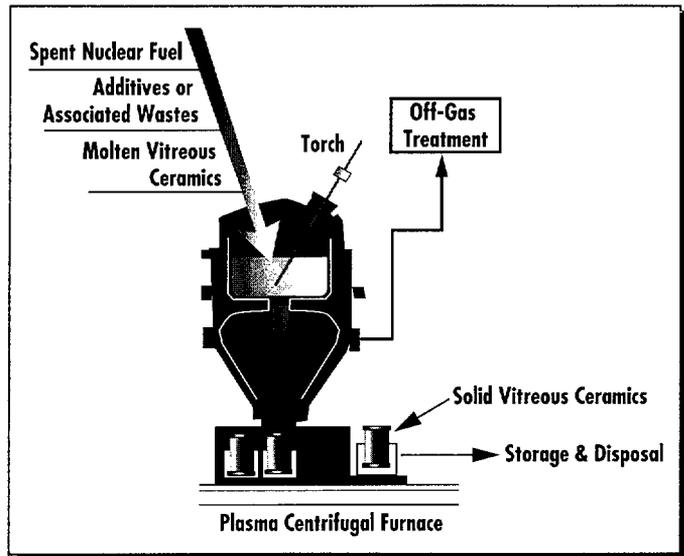
See cost summary sheet.

Other

General purpose process; flexible for other applications.

Plasma arc process operation is higher in temperature (1500-1600°C vs. 1000°C) than other melting processes.

Waste form qualification is required.



Treatment Technology Synopsis

Glass Material Oxidation and Dissolution System (GMODS)

Conceptual Description

A batch process in which the fuel and depleted uranium are melted in a glass melter. Lead dioxide is added to convert the metals to oxides and glass frit is added to make glass having an enrichment of 20%. The lead is removed and recycled.

Products

Approximately 800 borosilicate glass logs, packaged and configured for co-disposal. The glass logs are similar in number, configuration, uranium content and enrichment to the Dissolve and Vitrify option.

Secondary Streams

The melting process produces an offgas of volatile and semi-volatile fission products (e.g. cesium) that must be captured and disposed.

Metallic lead is produced and recycled.

Criticality Implications

The quantity being melted must be controlled to reduce the potential for in-process criticality. The lower enrichment and waste form stability reduces criticality concerns in repository.

Proliferation Resistance

The storage and disposal package contains LEU with no fission products removed. The glass form makes uranium recovery difficult.

Technical Maturity

Demonstrated in the laboratory, needs development. Production availability in 2006 time frame.

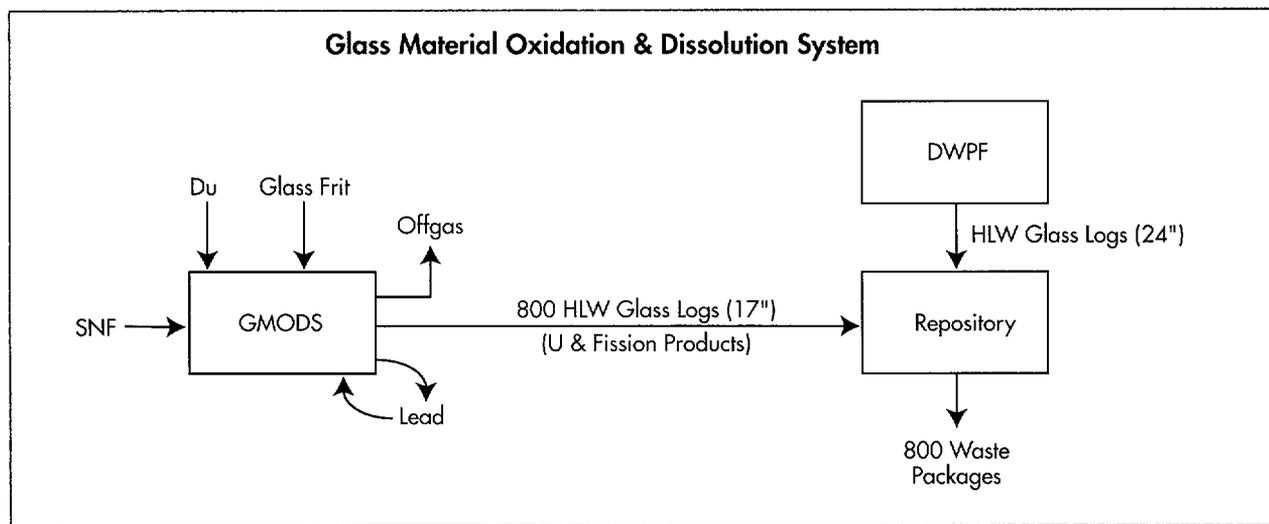
Cost

See cost summary sheet.

Other

General purpose process; flexible for other applications.

Waste form qualification is required.



Treatment Technology Synopsis

Dissolve and Vitrify

Conceptual Description

The fuel is dissolved in an acid dissolver and depleted uranium (DU) added to reduce U-235 enrichment to less than 20%. The dissolution product is fed directly to a vitrification plant, producing DWPF quality borosilicate glass cylinders sized for co-disposal. Uranium loading is limited to approximately 10 weight percent, to control glass quality.

Products

Approximately 800 glass logs, packaged and configured for co-disposal.

Secondary Streams

The melting process produces an offgas of volatile and semi-volatile fission products (e.g. cesium) that must be captured and disposed.

Criticality Implications

The quantity being dissolved must be limited to reduce the potential for in-process criticality. The 20% enriched uranium and waste form stability reduce criticality concerns in repository.

Proliferation Resistance

The storage and disposal package contains LEU with no fission products removed.

Technical Maturity

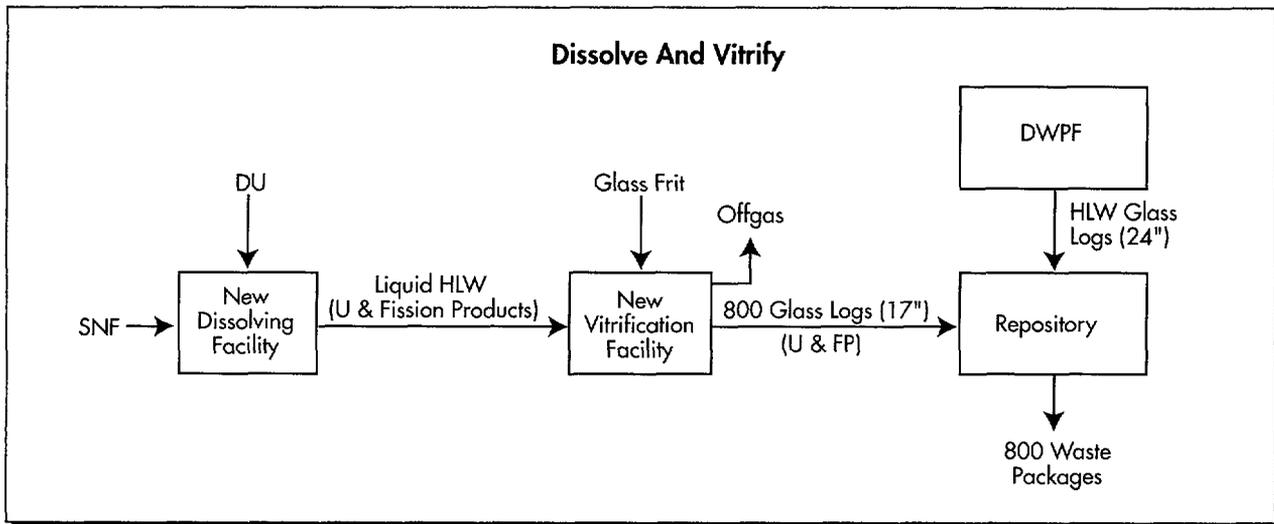
Both fuel dissolution and vitrification are state of the art. Adjustment of the molten feed to minimize vitrification process offgassing needs process development. Production availability in 2005-2010 time frame.

Cost

See cost summary sheet. Costs driven primarily by need for new dissolving and vitrification facilities.

Other

Waste form qualification is required.
Higher uranium loadings may be possible.



Treatment Technology Synopsis

Electrometallurgical Treatment

Conceptual Description

The fuel is cropped and melted, forming a metal ingot that is placed in an electrorefiner. The aluminum, uranium, and fission products are removed. Fission products are oxidized in a furnace and dissolved into glass. A second melter is used to dilute uranium with depleted uranium to less than 20% enrichment.

Products

Approximately 90 borosilicate glass logs, 2 ft. diameter by 10 ft. long.

Secondary Streams

The melting process produces an offgas of volatile and semi-volatile fission products (e.g. cesium) that must be captured and disposed.

Pure uranium, blended to low enrichment (<20%), suitable for recycling as commercial fuel. Aluminum disposed as LLW.

Criticality Implications

Waste form contains virtually no fissile material.

Proliferation Resistance

Fission products are separated from the uranium and the uranium is diluted to LEU.

Technical Maturity

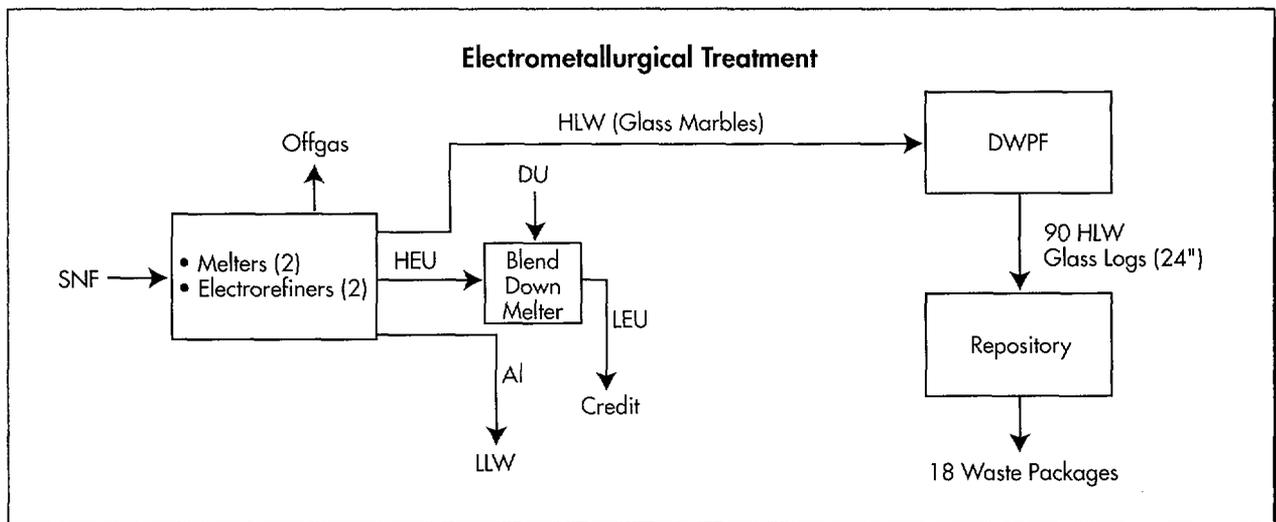
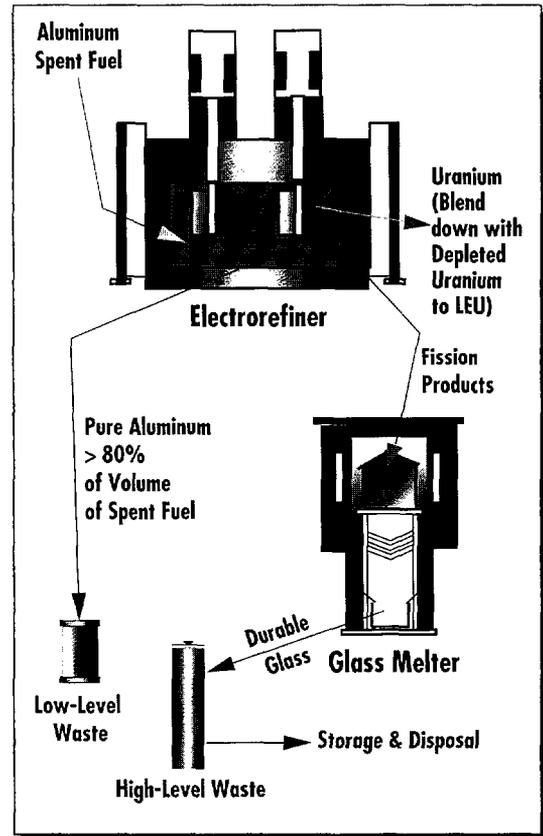
Process not demonstrated for aluminum-based SNF, but similar to process demonstrated at Argonne for EBR-II fuel. Uranium extraction has been proven. Aluminum extraction requires development. Production availability in 2005-2010 time frame.

Cost

See cost summary sheet.

Other

Process can be used for several fuel types but requires different head ends for each.



Treatment Technology Synopsis

Comparative Scenario: Processing/Co-Disposal

Conceptual Description

The fuel is dissolved in acid and the uranium is extracted and isotopically diluted for recycle (resale for commercial use). The waste products are fed to existing HLW tanks and converted to borosilicate glass logs. Processing through 2008 assumed; co-dispose the balance.

Products

Approximately 120 borosilicate glass logs, 2 ft. diameter by 10 ft. long.

Secondary Streams

- Offgas (internal to existing canyons)
- Saltstone, to LLW
- LEU

Criticality Implications

Waste material contains virtually no fissile material.

Proliferation Resistance

Fission products are separated from the uranium and the uranium is diluted to LEU.

Technical Maturity

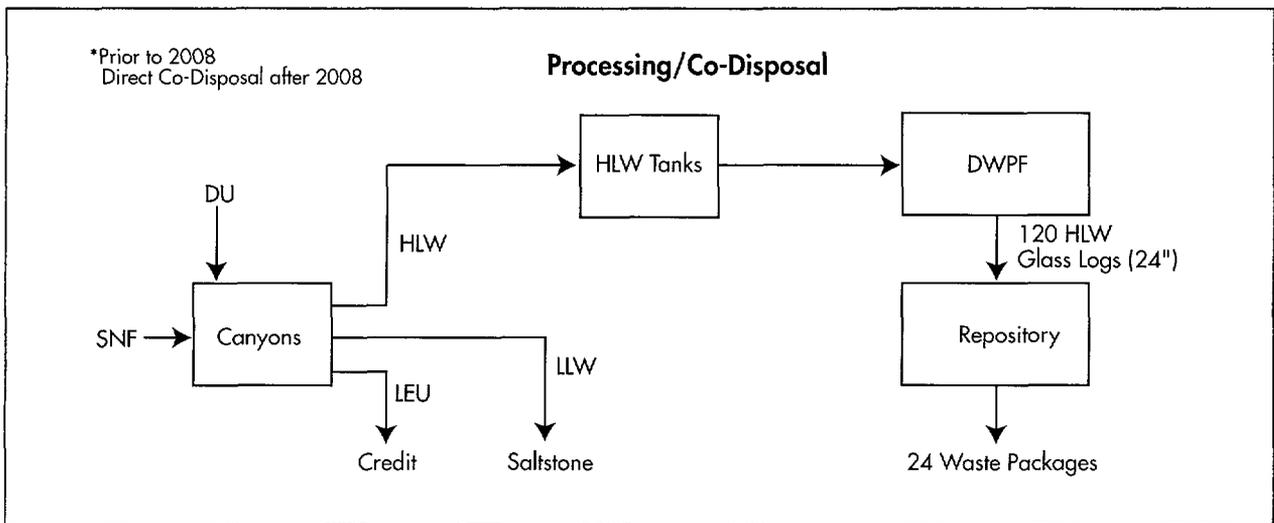
Process already in place; requires production change.

Cost

See cost summary sheet.

Other

This is the reference scenario for comparing the alternatives.
 Processing capability currently assigned to DNFSB 94-1 materials; availability exists after 2000 for additional SNF.
 Processing through 2008 assumed as a means to eliminate the inventory backlog.
 A different process is needed for SNF received after 2008. A likely method is Direct Co-Disposal.



Treatment Technology Synopsis

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4. Evaluating the Alternatives

Section 3 identifies alternatives and constraints regarding waste forms and packaging, fuel handling and storage facilities, and treatment technologies. The Team compared and evaluated the identified treatment alternatives to develop a basis for formulation of a technical strategy for the SNF in question. This section summarizes the evaluations and their conclusions.

The Team employed a structured approach for evaluating the treatment technologies that included both qualitative and quantitative methods. Sub-teams were assigned to examine factors such as waste form acceptability, handling and storage needs, treatment technologies, and cost and schedule projections. Experts were engaged to provide input and to review and comment on the Team's findings as they evolved. The Team established a framework for comparison of the alternatives, using common and consistently applied figures of merit to the greatest extent possible. Finally, a multi-attribute analysis (based on the Kepner-Tregoe method)¹¹ was conducted to provide a methodical, structured, and logic-based comparative evaluation.

These evaluations, and conclusions drawn from them, are described below.

4.1 Waste Form Characteristics and Performance

The SNF technologies under consideration produce waste forms with differing characteristics and expected behavior. In each case, the likelihood that its waste form will prove to be acceptable for disposal in the geologic repository is central to the viability of that technology. This section presents the Team's judgments regarding waste form acceptance for the technologies studied.

Waste Form Characteristics

The waste form characteristics of significance for repository disposal are listed in Table 4.1-1. These characteristics are extracted from the *Waste Acceptance System Requirements Document* (for high-level waste products)¹² and the *Preliminary Requirements for the Disposition of DOE Spent Nuclear Fuel* (for research reactor SNF).¹³

Treatment technologies were evaluated and compared from a waste form standpoint, considering each of the waste form characteristics in the table, and based on the conceptual level of information available in this study. The Team concluded that all of the treatment technologies could produce technically satisfactory waste forms. In many respects, the technologies produce comparable waste forms; however, there are several differences among them, as follows:

- Repository criticality control is dealt with in different ways. For the Direct Disposal technologies, limits (e.g., per Section 3.1) might be placed on the amount of U-235 permissible in each package. For other technologies, some combination of supplemental poison material or U-235 dilution (via addition of depleted uranium) might be employed.

Table 4.1-1 Summary of Waste Form Characteristics

Characteristic	Research Reactor SNF Waste Form
<p>Packaging</p> <ul style="list-style-type: none"> • Meets dimension and weight limits • Material compatibility • Thermal limits • Internal gas pressure limit • Labeling • Handling ability • Waste isolation 	<p>Treatment technologies produce satisfactory waste forms in all respects except gas generation. The Direct Disposal (including co-disposal packages) and Press and Dilute waste forms could cause hydrogen buildup due to aluminum oxidation; this would be controlled by the elimination of water and oxygen during staging, and venting prior to repository disposal.</p>
<p>Contents</p> <ul style="list-style-type: none"> • Solid material, no particulates • Non-combustible • No free liquids • Homogeneity • No RCRA waste 	<p>Treatment technologies produce waste forms which have the required characteristics, except:</p> <ul style="list-style-type: none"> • Potential hydrogen gas generation, as noted above. • Direct Disposal (including co-disposal packages) and Press and Dilute technologies do not produce homogeneous waste forms; homogeneity of Melt and Dilute and Plasma Arc technologies is not certain.
<p>Chemical Reactivity</p> <ul style="list-style-type: none"> • Not chemically reactive • Non-pyrophoric • Non-explosive 	<p>All of the technologies produce waste forms that have these characteristics.</p>
<p>Criticality Control</p>	<p>Criticality control is maintained for all technologies:</p> <ul style="list-style-type: none"> • For Direct Disposal (including co-disposal packages), by U-235 limitations per package. • For balance of treatment technologies, through HEU dilution (with depleted uranium) and/or neutron poison. • Electrometallurgical and Baseline (Processing) waste forms contain essentially no fissile material.
<p>Radiation</p> <ul style="list-style-type: none"> • Radiation field limits • Canister surface contamination limits 	<p>All of the technologies can produce waste forms that have these characteristics.</p>
<p>Safeguards</p>	<ul style="list-style-type: none"> • Waste forms are self-protecting (i.e., fission products are retained). • Direct Disposal (including co-disposal packages) and Press and Dilute contain mechanically separable HEU.

The degree of homogeneity of the poison/diluent in the mixture also differs, from a non-homogeneous layered arrangement (Press and Dilute), to a completely homogeneous mixture for the glass and ceramic waste forms. While these criticality control approaches require different analytical methods and involve some cost differentials (e.g., due to differing amounts and forms of poison/diluent required), they are all considered by the Team to be viable in concept.

- Hydrogen generation by radiolysis must be considered for those waste forms in which there is exposed aluminum metal (e.g., Direct Disposal and Press and Dilute). Concerns include pressure buildup in sealed canisters during interim storage and the potential for ignition of hydrogen in the presence of oxygen. In the Team's judgment, these are manageable and not likely to render any option unacceptable. The combination of effective drying and staging, and inerted well-sealed canisters minimizes gas generation. If necessary, the canisters can be vented prior to repository disposal to ensure that pressure limits within the waste package are not exceeded.
- The treatment technologies provide nuclear material safeguards in different ways. A dilution approach reduces SNF enrichment to <20% U-235 by melting or dissolving with depleted uranium. A self-protecting approach is provided by retaining fission products in the waste packages, and/or by storing them with other highly radioactive waste materials. Although mechanical or isotopic separation of the U-235 would be possible for several of the proposed waste forms, such separation would be difficult. The Team believes that all of the waste forms would be acceptable, given appropriate safeguards.

Waste Form Performance

Beyond the assessment of waste form characteristics, the Team also looked at the aggregate effect on radiological doses of disposing of this material in a geologic repository along with commercial SNF and HLW. A bounding assessment was conducted based on a scenario from the Total System Performance Assessment - 1995.¹⁴ This assessment estimated the fuel's contribution to the dose to the public at the accessible environment around the repository. The peak dose attributed to research reactor fuel is more than one order of magnitude below that from the peak dose of the entire repository inventory. (See Appendix E.)

While preliminary, this assessment recognized that research reactor fuel is a relatively small portion of the total SNF to be emplaced in the repository, and may have little or no influence on the repository's overall acceptability from an environmental and public health and safety standpoint.

■ 4.2 Cost Projections

Cost estimates were generated for each treatment technology. These estimates were developed primarily for use in comparing the technologies; for that reason, the Team placed high emphasis on achieving consistency of cost treatment and proper consideration of major cost drivers.

The estimates are also useful for decision makers in developing a preliminary sense of the overall costs that would be entailed in implementing one or more of the technologies. For this purpose, the Team developed cost uncertainty ranges which take into account the relative differences in maturity and complexity among the technologies, and the overall conceptual nature of this study.

The cost estimates were constructed in a methodical way, beginning with the identification of major cost components which together comprise the full costs of SNF handling, conditioning, packaging, storage, and disposal for each treatment technology. Common and consistent cost assumptions were agreed upon. Based on the assumptions, and employing information on costs of comparable projects or activities, the Team developed conceptual estimates for each cost component, for each technology. The Team then applied experience-based adjustments to these component conceptual estimates, to account for uncertainty. Finally, the adjusted component cost estimates were compiled to produce comparative costs (called "cost comparison points") for each treatment technology, along with an uncertainty band. In each case, net present value was also calculated to show the cost effect of projected schedule differences.

This cost estimating process, and its results, are explain in more detail in the following sections. Detailed information on the cost evaluation is provided in Appendix C.

■ ■ 4.2.1 Cost Assumptions

Several assumptions regarding the conceptual design and implementation of the treatment technologies are particularly influential on projected costs. For the technology cost estimates to be comparable, such assumptions must be consistently applied. Key examples follow.

New Versus Existing Facilities

It was assumed that existing SRS facilities and site infrastructure would be used whenever practical. New facilities were considered only when existing ones would be clearly inadequate. This assumption reduces cost estimates for the majority of the technologies. Cost treatment of secondary waste streams is an example. For technologies that produced liquid HLW streams, it was assumed that the existing SRS HLW tanks would be adequate for storage and disposal of the liquids.

Quantities and Timing of SNF Receipts

Handling, treatment, and packaging support for intra- and inter-site SNF transfers was estimated based on the capabilities of the existing SRS wet storage basins and the expected receipt schedules for centralizing all aluminum-based SNF at the SRS. Figure 4.2-1 displays the quantity and timing of expected SNF shipments to SRS. Additional intra-site shipments were assumed for SNF transfers in support of canyon operations, to permit use of the different capabilities at the RBOF and L-Basin, and for transfers of SNF from wet storage to interim dry storage.

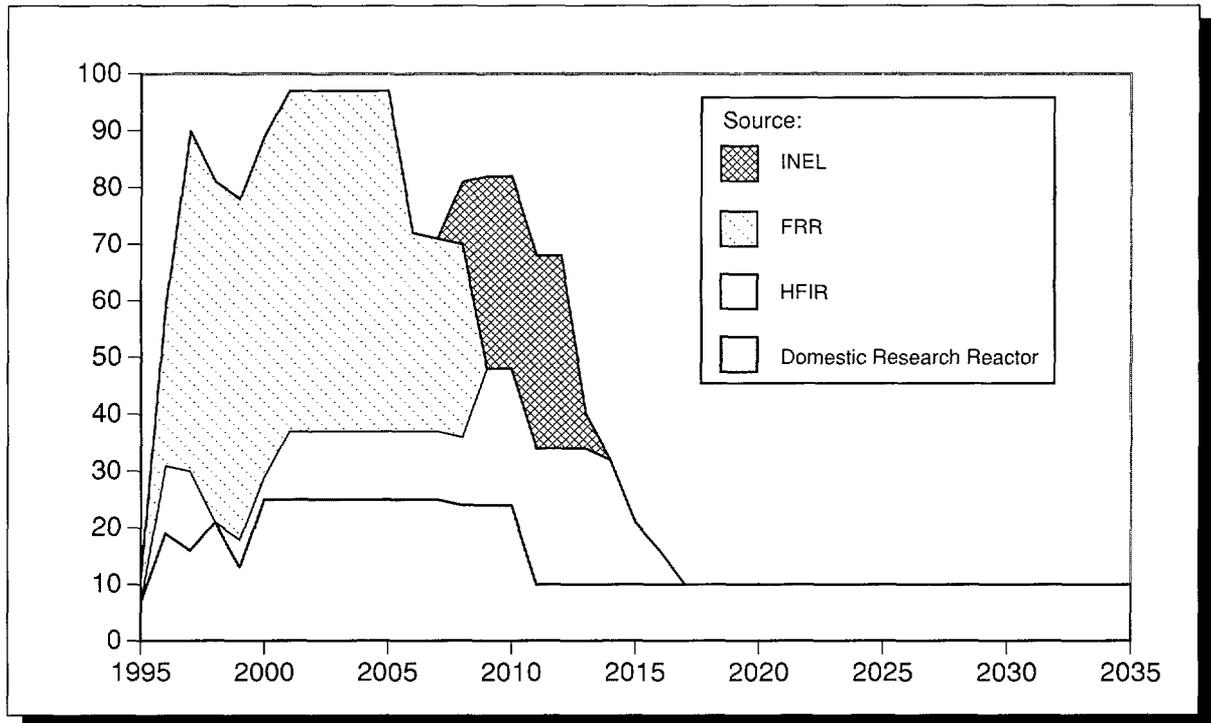


Figure 4.2-1 Expected Cask Receipt Rates

Schedule

Recognizing differences among the technologies in complexity, maturity, and need for new facilities, the Team evaluated for cost estimating purposes the relative timing and projected dates by which each could be available:

	<u>Startup Date</u>
• Direct Disposal / Co-Disposal Packages	2001
• Press and Dilute, Melt and Dilute	2003
• Electrometallurgical Treatment	2005
• Plasma Arc, GMODS, Dissolve and Vitrify	2006

These dates are aggressive. In each case, they are based on the assumption that traditional appropriations, budgeting and management practices will be significantly compressed to achieve success.

Also, it was assumed that once treatment operations begin, off-site SNF receipts would be sent directly to the transfer facility, and that on-site SNF would be transferred from wet storage as quickly as the treatment capacity permits. The objective would be to de-inventory the wet basins within six or seven years in order to minimize operating costs. Beyond that point, there would continue to be significant annual SNF throughput until the last INEL SNF is received around 2012. While SNF receipts continue indefinitely, they are significantly reduced after 2012. Operations (including treatment) at the transfer facility can then be scaled back for a few years until the repository begins operation.

It was assumed that SNF inventory in interim storage would be shipped to the repository over an eight year period beginning in 2020. Shipments after that would be made as needed to keep up with incoming receipts from operating reactors.

■ ■ 4.2.2 Conceptual Cost Estimates

Based upon the above, the Team developed conceptual costs by category for each technology. The estimates were drawn from experience in comparable applications, from third-party input, from simple estimating calculations, and from the Team members' judgment. These are discussed below.

Wet Storage and Handling

This component includes costs associated with off-site receipts and the use of existing wet basins at SRS prior to SNF treatment and interim storage. They are primarily O&M costs along with minimal capital expenditures. The cost estimates for this component are influenced primarily by the length of time during which RBOF and L-Basin would be needed. The later startup dates projected for the advanced treatment technologies dictate longer periods of basin usage and attendant higher costs. A much smaller cost element within this component is that associated with the receipt of off-site SNF (Figure 4.2-1), and is essentially the same for all technologies.

Direct Disposal	Direct Co-Disposal	Press & Dilute		Melt & Dilute	Plasma Arc	GMODS	Dissolve & Vitrify	Electrometallurgical	Processing/Co-Disposal
		20%	2%						
280	280	350	350	350	460	460	460	440	430

Transfer and Packaging

These are pre- and post-treatment handling costs for moving the SNF to the treatment facility and packaging the waste form prior to interim storage. The cost of a dry transfer facility is included. The cost estimate for this new facility is based on two INEL reports¹⁵ which examined SNF handling and facilities costs.¹ For the comparative scenario, direct disposal of SNF after 2008 was assumed to require a much smaller transfer facility.

Direct Disposal	Direct Co-Disposal	Press & Dilute		Melt & Dilute	Plasma Arc	GMODS	Dissolve & Vitrify	Electrometallurgical	Processing/Co-Disposal
		20%	2%						
440	430	420	440	390	380	390	390	360	170

Treatment

Treatment costs are those directly associated with application of the treatment technology to produce a waste form acceptable for interim storage and repository disposal. This component includes the capital costs for the treatment facility and equipment, and the operations costs required for treatment of the SNF. Estimated costs were based in part on input from advocates for each technology, with experience-based judgements by the Team to achieve consistency.

¹ Because of the similarity of functions performed and production capacity required, Module U from these references was used as the basis for capital costs, and decontamination and decommissioning costs of the dry transfer facility. Systems that exceeded requirements for research reactor SNF operations were scaled back.

For the Direct Co-Disposal technology, treatment is limited to vacuum drying the SNF; costs for vacuum drying are included in the Transfer and Packaging component above. For the Dissolve and Vitrify technology, the Team considered using the DWPF as the treatment facility. However, due to criticality issues associated with the HEU content of the SNF, it was decided that a new vitrification facility would be needed for this option.

Treatment Costs (\$M)									
Direct Disposal	Direct Co-Disposal	Press & Dilute		Melt & Dilute	Plasma Arc	GMODS	Dissolve & Vitrify	Electromet-allurgical	Processing/Co-Disposal
		20%	2%						
0	0	230	230	270	450	410	720	600	640

Interim Storage

Cost estimates for interim dry storage were based on a common conceptual modular dry vault configuration and cost model, scaled according to the number of canisters produced for each technology. The two INEL reports noted above provide the basis for the capital, O&M, and decontamination and decommissioning costs for interim dry storage. For economy of operations, it was assumed that interim storage modules would be constructed as add-ons to the transfer facility.

Interim Storage Costs (\$M)									
Direct Disposal	Direct Co-Disposal	Press & Dilute		Melt & Dilute	Plasma Arc	GMODS	Dissolve & Vitrify	Electromet-allurgical	Processing/Co-Disposal
		20%	2%						
120	130	100	120	100	90	110	110	0	50

Disposal

This component includes costs required for transport of the treated waste forms to the repository and final packaging and emplacement at the repository.

Transportation costs are directly proportional to the number of shipments required. Unit costs were developed for several shipping package designs (for the different interim storage canisters) and then multiplied by the number of waste packages.

Disposal Costs – Transportation (\$M)									
Direct Disposal	Direct Co-Disposal	Press & Dilute		Melt & Dilute	Plasma Arc	GMODS	Dissolve & Vitrify	Electromet-allurgical	Processing/Co-Disposal
		20%	2%						
40	40	20	40	20	20	30	30	10	20

A prorated share of the overall repository development costs was included to cover repository characterization, site preparation, and licensing for this SNF. A cost of \$30 million was assumed for all technologies, based on the aggregate quantity of SNF to be disposed.

Disposal Costs – Repository Development (\$M)									
Direct Disposal	Direct Co-Disposal	Press & Dilute		Melt & Dilute	Plasma Arc	GMODS	Dissolve & Vitrify	Electrometallurgical	Processing/Co-Disposal
		20%	2%						
30	30	30	30	30	30	30	30	30	30

Disposal Operations costs are incurred following receipt of the SNF canisters at the repository. These are directly proportional to the number of repository waste packages and include loading of the interim storage canisters into the waste packages and emplacement in the repository drift.

Disposal Costs – Operations (\$M)									
Direct Disposal	Direct Co-Disposal	Press & Dilute		Melt & Dilute	Plasma Arc	GMODS	Dissolve & Vitrify	Electrometallurgical	Processing/Co-Disposal
		20%	2%						
370	140	40	130	40	40	80	80	10	40

Adjustments

This last component was included to identify special additional costs or credits, as applicable. For the direct disposal and dilution technologies, an additional cost is assumed for the conditioning of the SNF (i.e., metallic uranium fuels) for which cold vacuum drying (already included) may not be adequate to achieve waste form acceptability. An additional cost of \$60M is estimated for hot vacuum drying. An alternate to conditioning these fuels would be to process them; the cost for processing is estimated at \$30M.

For the Electrometallurgical Treatment technology and the comparative scenario, credit is given for the value of the uranium available for resale into the commercial market.

Adjustments (\$M)									
Direct Disposal	Direct Co-Disposal	Press & Dilute		Melt & Dilute	Plasma Arc	GMODS	Dissolve & Vitrify	Electrometallurgical	Processing/Co-Disposal
		5	2%						
60	60	60	60	0	0	0	0	-220	-180

4.2.3 Cost Uncertainty

Preliminary cost estimates of large, complex, long term projects are inherently uncertain. To account for uncertainty, the Team developed a cost range, spanning the conceptual estimate for each technology. The Team also considered the relative uncertainty among technologies, and applied adjustment factors to account for the relative uncertainty in the cost comparison. [See sections 4.3 and 4.4 for a discussion of the overall technology comparison.]

For each technology, the cost band shown on Figure 4.2-2 represents the Team's judgement as to the range within which a more complete and better developed cost projection, once available, would likely fall. To estimate this range, the Team considered the source of the cost data and the maturity of the technologies. In each of the cost components of each technology, the conceptual estimate was assessed a negative (-%) and a positive (+%) variance. These variances were then "rolled up" by using the negative variances to calculate the low end of the range and the positive variances to calculate the high end. Details on the cost variances are presented in Appendix C.

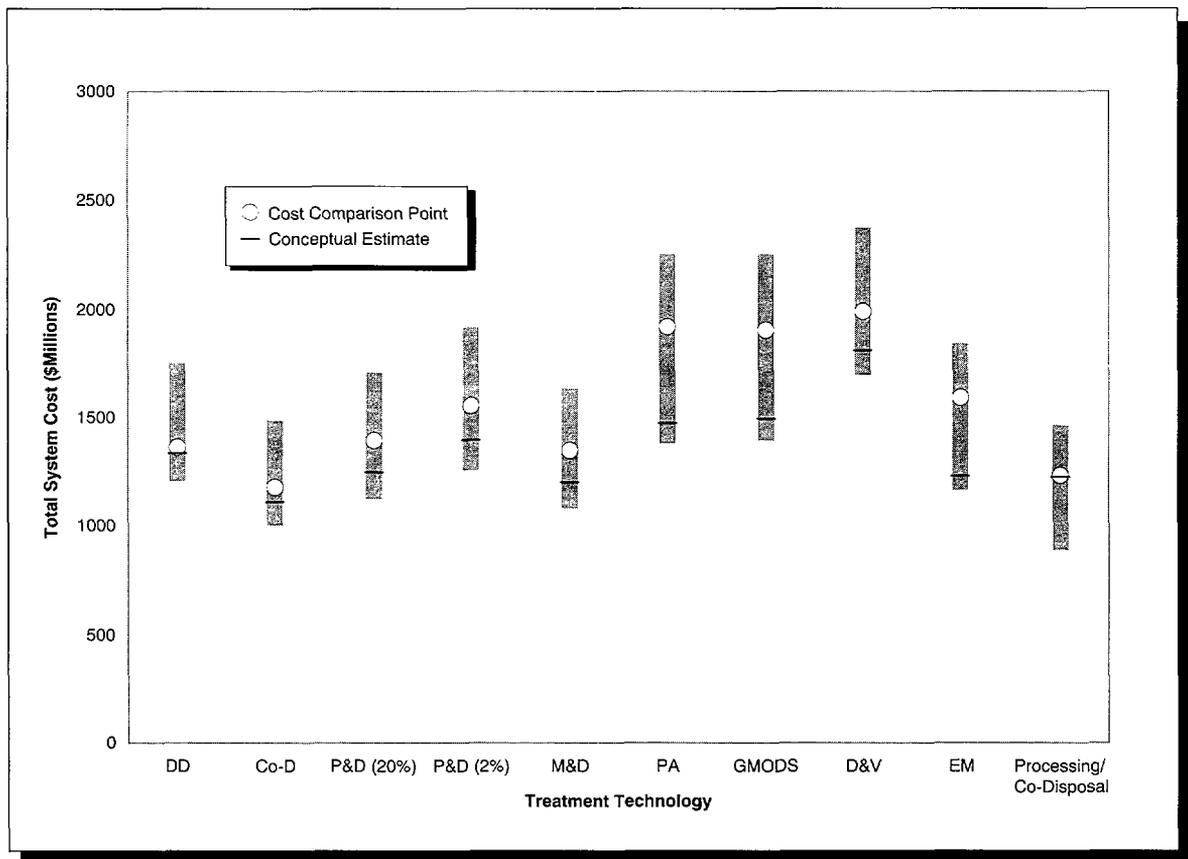


Figure 4.2-2 Range of Cost Estimates

The degree of uncertainty varies among the different technologies. Some are very conceptual, and their cost estimates are considered more uncertain relative to other technologies. A simple measure of this relative uncertainty is the difference among the uncertainty variances discussed above. Table 4.2-1 lists "Relative Uncertainty Factors" which are derived directly from the uncertainty variances in Appendix C.

As examples of relative uncertainty, consider the wet storage and handling cost component and the treatment cost component.

- Based on historical data, the variance range for wet storage and handling costs was assumed to be the same for all the technologies. Because the wet storage and handling activities are the same for all, no relative uncertainty exists.
- On the other hand, there are substantial differences in uncertainty among treatment technologies due to the significant differences in maturity and complexity. The comparative scenario, Processing/Co-Disposal, was considered the most certain. Simple technologies (Direct Disposal and Dissolve and Vitrify) were close to the base; more advanced technologies (Plasma Arc and GMODS) were further from the base and have higher relative uncertainty factors.

Table 4.2-1 Relative Uncertainty Factors

Option	Cost Category						
	Wet Storage and Handling	Transfer and Packaging	Treatment	Interim Storage	Disposal		
					Transportation	Repository Development	Disposal Operations
Direct Disposal	None	Base		+20%	None	None	None
Direct Co-Disposal		+10%	+10%				
Press and Dilute (20%)			+40%				
Press and Dilute (2%)							
Melt and Dilute			Base				
Plasma Arc				+90%			
GMODS				+20%			
Dissolve and Vitrify		+60%					
Electrometallurgical							
Processing/ Co-Disposal	None	Base	Base	Base / +20% (Note 2)	None	None	None

- Notes:
1. In each cost category, the technology assigned the least uncertainty is shown as the "base". If all technologies within a category were considered to have similar uncertainty values, then no relative uncertainty factor is shown.
 2. A +20% uncertainty factor is shown for interim storage of the SNF to be treated by direct disposal (co-disposal packages) after 2008.

■ ■ 4.2.4 Cost Comparison

Table 4.2-2 lists the individual cost component estimates, the relative cost uncertainty adjustment, and the total “cost comparison point” for each technology. Cost comparison points were calculated by summing the cost component conceptual estimates, as adjusted for relative uncertainty. Figure 4.2-2 also shows these cost comparison points

In the judgement of the Team, the cost comparison point – essentially a conceptual cost estimate adjusted for relative uncertainty – is the most meaningful basis for cost comparison of the technologies within the limitations of this study.

Finally, the Team calculated the net present value of the total system costs for each technology, based on the cost comparison points and assumed facilities construction and operations schedules. These estimates are provided in Table 4.2-3.

Table 4.2-2 Cost Estimate Summary
(Millions of 1996 Dollars)

Option	Conceptual Costs by Category ¹							Relative Cost Uncertainty ²	Cost Comparison Point ³	
	Wet Storage and Handling	Transfer and Packaging	Treatment	Interim Storage	Disposal					Adjustments
					Transportation	Repository Development	Disposal Operations			
Direct Disposal	280	440	0	120	40	30	370	60 (Note 4)	20	1400
Direct Co-Disposal	280	430	0	130	40	30	140	60 (Note 4)	70	1200
Press and Dilute (20%)	350	420	230	100	20	30	40	60 (Note 4)	150	1400
Press and Dilute (2%)	350	440	230	120	40	30	130	60 (Note 4)	160	1600
Melt and Dilute	350	390	270	100	20	30	40	0	150	1300
Plasma Arc	460	380	450	90	20	30	40	0	440	1900
GMODS	460	390	410	110	30	30	80	0	410	1900
Dissolve and Vitrify	460	390	720	110	30	30	80	0	180	2000
Electrometallurgical	440	360	600	0	10	30	10	-220 (Note 5)	400	1600
Processing/ Co-Disposal	430	170	640	50	20	30	40	-180 (Note 5)	10	1200

- Notes
1. Estimates are based on conceptual studies, without contingency or uncertainty adjustments, and are rounded to the nearest \$10M.
 2. Adjustment for relative uncertainty among technology options, per section 4.2.3.
 3. The adjusted cost on which the comparative evaluation of options is based. See sections 4.2.4 and 4.3. These costs are rounded to the nearest \$100M.
 4. Charges for a conditioning (hot vacuum drying) facility for fuels which require that treatment for direct disposal. If instead these fuels are processed, the adjustment is reduced to \$30M to reflect the cost of processing.
 5. Credit for sale of U-235.

Table 4.2-3. Present Value of Total System Costs
(Millions of 1996 Dollars)

Technology	Total System Costs (Cost Comparison Point)	Net Present Value of Total System Costs
Direct Disposal	\$1,400	\$900
Direct Co-Disposal	\$1,200	\$800
Press and Dilute (20%)	\$1,400	\$1,000
Press and Dilute (2%)	\$1,600	\$1,100
Melt and Dilute	\$1,300	\$1,000
Plasma Arc	\$1,900	\$1,300
GMODS	\$1,900	\$1,300
Dissolve and Vitrify	\$2,000	\$1,300
Electrometallurgical	\$1,600	\$1,100
Processing/Co-Disposal	\$1,200	\$800

■ 4.3 Treatment Technologies Compared

The information and projections for each of the treatment technologies were compared on a common basis, by compilation in a single matrix (Table 4.3-1). The information is necessarily cryptic, but is nonetheless useful in presenting a broad and balanced comparison of these very different technologies. The data are arranged in two main categories, as follows:

Technical Performance

This is a summary of each technology's effectiveness and relative merits as a disposal methodology. Information is provided as to waste form produced, number of packages and configuration, and how the technology deals with criticality and proliferation.

Implementation

This is a summary of those factors related to the expected degree of difficulty in developing, demonstrating, engineering, constructing and licensing the systems and facilities required for each technology. Information is provided as to technical maturity (development required), new facility requirements, timing and costs.

Table 4.3-1 Technology Comparison Summary

Option	Technical Performance				Implementation				Other Factors
	Waste Form	Packaging Configuration	Disposal Package Criticality	Proliferation	Development	New Facilities	Timing	Cost (\$B)	
Direct Disposal	SNF assemblies packaged and dried	~ 1100 canisters suitable for direct disposal	HEU (as high as 93%) per package U-235 mass planning value	Self protecting, no fission product removal	Minimal system development Waste form qualification required	Transfer Facility (with packaging & drying systems) Interim Storage	FY98 - start design & build project FY01 - start packaging & drying operations	1.4	Requires early NRC concurrence for disposal of HEU Transfer Facility may be eliminated by extending wet basin operations or by prepackaging at reactor sites.
Direct Co-Disposal		~ 1400 canisters configured for co-disposal with HLW glass logs						1.2	
Press & Dilute	SNF compacted and diluted (2 or 20% enrichment)	~400 co-disposal canisters (20% enrichment)	Enrichment reduced to 20% or 2% by layering	Void spaces reduced U-235 mass planning value	Moderate system development Waste form qualification required	Transfer Facility (with pressing, drying & packaging systems) Interim Storage	FY99 - start design & build project FY03 - start operations	1.4 (20%)	Requires early NRC concurrence for disposal of HEU Powder fuels require additional treatment.
Melt & Dilute		~1300 co-disposal canisters (2% enrichment)	Enrichment reduced					U-235 dilution	
Plasma Arc	Vitreous ceramic (20% enrichment)	~400 co-disposal canisters	Homogeneous configuration eliminates void spaces U-235 mass planning value	Self protecting, no fission product removal	Substantial system development Waste form qualification required	Transfer Facility (with melting & packaging systems), Interim Storage	FY01 - start design & build project FY06 - start operations	1.9	
GMODS	Borosilicate glass logs (20% enrichment)							~800 co-disposal canisters	
Dissolve & Vitrify	Borosilicate glass logs Uranium diluted for commercial sale	~90 HLW glass logs	No uranium	No uranium in final waste form Separated uranium is isotopically diluted	Moderate system development Waste form is qualified	Transfer Facility (with hot cell for EM process) Augmented GWSB	FY00 - start design & build project FY05 - start operations	1.6	Uses existing DWPF systems for HLW Separated uranium re-cycled for use in commercial fuel
Processing/Co-Disposal								~120 HLW glass logs and ~300 direct co-disposal canisters	
					Balance same as direct disposal	Interim Storage	After FY08 - direct dispose		After 2008, direct co-disposal

Considered in cost?

■ 4.4 Multi-Attribute Evaluation

This evaluation was essentially a Kepner-Tregoe (K-T) multi-attribute analysis,¹⁶ tailored to fit this case. It is important to note that for an evaluation of this kind – particularly one in which the alternatives being evaluated are fundamentally different and have large uncertainties – the process provides a logical structure for comparison and valuable insights into the relative merits of the alternatives. It is not, however, “quantitative” in any real sense. The numerical weights and scores make it possible to compare judgments, but they do not have absolute value. Properly used, this kind of analysis is one evaluation tool – not a way to calculate the “answer.”

The Approach

As a first step, a set of evaluation categories and criteria were established by which each technology could initially be screened and then ranked against each of the other technologies. The screening review served to eliminate from further consideration technologies which would face significant technical, implementation, or scheduling hurdles and therefore were considered to have significantly lower prospects for success in treating aluminum-based SNF.

Table 4.4-1 lists the screening and evaluation criteria. The bases for the screening criteria are as follows:

- DOE requires the necessary development work to select an alternative technology to be completed before 2000, so that if a technology cannot be successfully developed by this time, the SNF can be processed in the operating Savannah River canyon. It is expected that other higher priority materials will be processed to the point that the canyon capacity will be available for spent fuel processing starting in the year 2000.
- Funding required during the development period (e.g., first 5 years) must be within that reasonably expected to be available in that time frame.
- The waste form must be compatible with anticipated repository requirements.
- The treatment technology cannot present any operational concerns such that environment, safety, and health requirements cannot be met.

Any alternative which, in the judgment of the Team, would not meet all of the above “must” criteria was eliminated from further consideration. (Three alternatives were rejected on the basis of screening requirements; these are described in Section 3.3.4.)

Following screening, a more detailed evaluation of the remaining alternatives was conducted. For this evaluation, a detailed set of decision factors and criteria was developed. As shown in Table 4.4-1, four evaluation categories were identified with each category providing a basis for comparison among the technologies. These are:

Table 4.4-1. Screening and Evaluation Criteria

Evaluation Category	Screening Criteria	Weighting Factor	Evaluation Scale
<p>Confidence in Success Evaluators' confidence that the technology can meet technical performance requirements, at predictable cost and schedule, based on such factors as:</p> <ul style="list-style-type: none"> • application of proven technology • waste form acceptance • licensing precedent • need for prototype • concept technical maturity (e.g., conceptual design, preliminary engineering) • basis for cost/schedule prediction • process simplicity 	<p>Technology development must be reasonably completed by 2000.</p>	<p>30</p>	<p>1 - 10 scale reflecting evaluators' judgment of the success likelihood of each candidate – higher confidences yield higher scores</p>
<p>Cost Projected total implementation costs including:</p> <ul style="list-style-type: none"> • life cycle cost (net present value) • near term (10 year) costs <p>Total implementation costs include R&D, licensing, design, construction, O&M, decommissioning, waste stream disposal, etc.)</p>	<p>Costs must be less than \$500 million for the first 5 years</p>	<p>30</p>	<p>1 - 10 sliding scale, higher cost options receive lower scores</p>
<p>Technical Suitability Merits of each technology with respect to:</p> <ul style="list-style-type: none"> • environment, safety, and health standards (ES&H) • waste form compatibility with repository requirements • nonproliferation • other missions requirements 	<p>Technology will be considered only if: Final waste form is compatible with anticipated repository disposal requirements. Meet current worker, public and environment, Environmental, Health and Safety requirements</p>	<p>20</p>	<p>1 - 10 sliding scale reflecting evaluators' judgment, taking into account such factors as waste packaging efficiency, reliance on package for long term protection, chemical stability, proliferation vulnerability, etc.</p>
<p>Timeliness Projected time to implement technology, (including R&D, licensing, engineering, procurement, construction, demonstration, startup, etc.)</p>		<p>20</p>	<p>1 - 10 scale with the highest score (10) assigned to any technology projected to be in place and operating by 2001. Score is reduced for startups beyond that date.</p>

- **Confidence in Success** - the degree to which the Team felt that each concept could be successfully executed, delivering a technically satisfactory, licensable waste product on time and within cost projections. This category was included to incorporate real-world practicality into the comparison process, ensuring that vulnerabilities (i.e., potential difficulties in resolving technical problems or in qualifying a waste form) were considered.
- **Cost** - a measure of the total system costs of the alternatives, on a life-cycle basis, as presented in Table 4.2-3.
- **Technical Suitability** - the degree to which each alternative was judged by the Team to be a technically attractive solution, in terms of waste form stability, proliferation resistance, operational simplicity, storage efficiency, etc. It was assumed that all options being evaluated (i.e., which passed the screening criteria) satisfied minimum technical and regulatory requirements – but that potentially significant technical differences should be taken into account.
- **Timeliness** - a measure of the projected time that would be required for each of the alternatives to complete the development, design, construction and testing phases, to the point of full operation.

Clearly the above categories are interrelated. For example, the simpler system alternatives are likely to have both shorter schedules and lower costs. But they were considered by the Team to be sufficiently distinct and important to warrant separate evaluation and comparison. Weights were assigned to each category, as listed on Table 4.4-1, reflecting the Team's judgment as to their relative importance.

Each of the treatment technologies was scored in each evaluation category, using a 1 to 10 rating scale. Scores were assigned in Team meetings, with substantial discussion and debate accompanying each. Technology advocates were consulted and their input was given careful consideration. In their final form, the scores reflect the consensus view of the Team.

Two points are particularly important regarding this evaluation process and conclusions which may be drawn from it:

- The approach considers primarily *relative*, rather than absolute, projected effectiveness of the technologies. For example, as described in Section 4.2 above, the cost projections are based in large part on conceptual information, with significant uncertainties. These costs can not be considered accurate forecasts of actual project financial requirements – but are legitimately useful for comparing various alternatives.
- The weightings and scores represent, in large measure, *qualitative* judgments of the evaluators. Assignment of numerical values (i.e., the K-T method) permits methodical comparison and compilation of results, but the comparisons remain qualitative in nature.

The Results

Table 4.4-2 shows the individual scores and the compiled results. The relative scoring of the alternatives in each of the evaluation categories is summarized below:

- **Confidence in Success:** The primary differentiators in this category were the relative maturity of the technologies (i.e., degree to which they could be considered “proven technology”) and the likely acceptability of the waste forms produced. The direct disposal and dilution alternatives generally scored well in the former and not as well against the latter; the inverse was true of the advanced technologies.
- **Cost:** Cost scores are a direct reflection of the life cycle costs shown on Table 4.2-3. The simpler technologies with shorter schedules scored higher than the advanced technologies.
- **Technical Suitability:** Repository waste form considerations were the primary differentiators in this category, with waste forms most like those currently anticipated for commercial SNF and vitrified HLW (e.g., low enriched, vitrified forms) receiving the highest scores. Scores were also influenced by technical complexity of the alternatives, with the simpler approaches rated more highly.
- **Timeliness:** This score correlates with the schedules discussed in Section 4.2.1. The simpler technologies received higher scores, on that basis.

All of the above factors contribute to the overall scores, as shown on Table 4.4-2. Some conclusions are evident:

- There appears to be clear preference for the simpler approaches – that is, the direct disposal and dilution technologies – compared to the advanced technologies, as indicated by the distinct families of overall scores.
- Among the simpler options, the Direct Co-Disposal approach was the clear preference of the Team.
- The other simple approaches (Direct Disposal in Small Packages, Press and Dilute, and Melt and Dilute) received essentially the same overall scores and should be considered equivalent, as evaluated here.
- Of the advanced treatment options, Electrometallurgical Treatment scores the highest. The most significant factor in the choice of this option over the others was the degree of actual experience with processing SNF.

Table 4.4-2. Evaluation Results

Technology	Confidence in Success (30%)	Cost (30%)	Technical Suitability (20%)	Timeliness (20%)	Overall Score
Direct Disposal	5	7	4	8	60
Direct Co-Disposal	6	10	5	8	74
Press & Dilute (20%)	6	7	7	6	63
Press & Dilute (2%)	7	5	6	6	62
Melt & Dilute	6	8	7	6	66
Plasma Arc	1	2	6	1	23
GMODS	1	2	7	1	25
Dissolve and Vitrify	4	1	7	1	31
Electrometallurgical	3	5	8	3	46

* currently selected 10/13/98

Sensitivity Study

A multi-attribute method such as the one used here is also useful for sensitivity analysis, as a means of challenging assumptions and better understanding the results. In this case, a very simple sensitivity check was performed by removing from consideration, in turn, each of the four major evaluation categories, and then recalculating and comparing the total scores. The results are tabulated in Table 4.4-3.

As shown in the table, the rankings of the alternatives are generally the same in each case – that is, the overall comparative analysis results are not dominated by any one assumption, criterion or evaluation. For example, in three of the four sensitivity cases, Direct Co-Disposal is the highest ranking alternative, and in the fourth case it ranked second among all options. The simple alternatives (the direct disposal and dilution approaches) ranked higher than the advanced technologies in all cases, and the Electrometallurgical Treatment technology was the highest ranking of the advanced options in all cases.

This simple sensitivity study provides a measure of confidence that the overall evaluation results are sound. Again, however, the process employed here is fundamentally a matter of tabulating judgments — in the final analysis, the results are only as valid as the judgments of those conducting the evaluation.

Table 4.4-3. Sensitivity Study Results

Technology	Base Ranking (With all evaluation categories included)	Sensitivity Case Rankings			
		Case 1 (with Confidence in Success excluded)	Case 2 (with Cost excluded)	Case 3 (with Technical Suitability excluded)	Case 4 (with Timeliness excluded)
Direct Disposal, Small Packages	5	3	5	3	5
Direct Co-Disposal	1	1	2	1	1
Press & Dilute (20%)	3	3	3	4	3
Press & Dilute (2%)	4	5	1	5	4
Melt & Dilute	2	2	3	2	2
Plasma Arc	9	8	9	8	9
GMODS	8	7	8	8	8
Dissolve and Vitrify	7	9	7	7	7
Electrometallurgical	6	6	6	6	6

5. Assembling the Path Forward

In Section 4, the alternative treatment technologies were evaluated essentially on a “stand-alone” basis; that is, each option was evaluated as if it were to be employed in treating the entire inventory of SNF. This is an artificial but useful simplification for comparing the treatment technologies, in that it minimizes the number of comparative cases and it highlights relative advantages and disadvantages which might be less apparent in a more complicated comparative model.

In actual practice, however, it is very unlikely that a single treatment approach would prove optimal for all of the fuel. Clearly, each of the treatment options is better suited for some fuel types than for others and, for some treatment options, universal application would be essentially impossible. For that reason, a composite or “hybrid” approach is the most likely success path for the SNF in question. Such an approach could include selective processing for those fuel types most suitable, followed by the application of one or more of the available technologies to the remaining SNF based on cost, schedule and other considerations.

The following sections address the development of this composite approach, including issues to be resolved and the implementation steps to be taken.

■ 5.1 The Composite Approach

From the evaluation described in Section 4, the Team concluded that all of the technologies considered would be capable of converting the SNF to a disposable form. None, however, seems optimal for all of the fuel types, and none is considered sufficiently mature to justify exclusive reliance by DOE. Therefore it stands to reason that at this point the DOE course of action should involve further examination and development of several parallel paths to provide maximum flexibility and to improve the overall success potential.

In selecting the recommended set of technologies for further action, the Team applied the following criteria:

1. The field should be narrowed substantially from the present group of treatment technologies.
 - Several of the technologies evaluated seem clearly preferable to others, based on the comparative evaluation.
 - Successful development programs require focus; too many simultaneous activities would drain resources and attention from the most promising avenues.
 - Similarities exist among several of the technologies, such that pursuing all would automatically entail certain redundant activities.
2. The selected set of technologies should be balanced and diverse, in the sense that there should be no single vulnerability common to all.

3. The evaluation showed clearly the benefits – cost, schedule, technical success potential – of the simpler treatment and disposal technologies. The selected set of options, and the implementing actions, should focus primarily on a simple technological approach.
4. The disposal concept of utilizing vitrified waste canisters for geologic disposal of SNF (i.e., co-disposal packages) is clearly attractive. The set of technologies pursued should be amenable to this concept.

Based on these criteria, the Team established the basic framework for a composite approach, as follows:

- **Direct Disposal in Co-Disposal Packages** is recommended as the primary approach, deserving lead status and full DOE support. This is the simplest of the technology options evaluated, and seems technically achievable in all respects, at moderate cost and on a timetable consistent with DOE's needs. Its single major vulnerability – the uncertain acceptability of HEU in the repository – argues for substantial analytical and regulatory effort in the near term to resolve this concern.
- **Press and Dilute, Melt and Dilute**, or some variation of these (as selected by further evaluation) is recommended as the dilution option to be developed in parallel with Direct Co-Disposal. This approach is also simple and scored very well in the comparative evaluation. Its waste form (in which HEU is blended with depleted uranium to produce a composite product with an effective enrichment in the LEU range) is potentially more acceptable and more easily licensed than would be direct disposal of HEU. It is fully compatible with the co-disposal packaging strategy.

These two approaches should be carried further in development, in parallel, over the near term (several years), each with specific development objectives. These parallel paths will lead to a scheduled decision point as to treatment and disposal of all of the research reactor SNF types. That final path could include one, or both treatment technologies, as dictated by the results of the development effort.

- A third option, Electrometallurgical Treatment, is recommended as an additional backup. This approach is fundamentally different from the others and, in that sense, offers diversity, flexibility, and protection from unforeseen technical or licensing difficulties. Its waste form – borosilicate glass logs, much like that to be produced by the DWPF – is very robust and highly likely to meet the regulatory requirements to be imposed. It offers the additional advantage of flexibility and potential multi-use.

DOE can follow this work as it is applied to non-aluminum fuel treatment. If the need for application of this technology should arise, then funds can be spent to adapt it for use with aluminum-based SNF. In the meantime, funding should be concentrated on development of the primary technologies.

The implementation of this parallel, converging development effort is discussed in more detail in Section 5.3.

■ 5.2 Selective Application of Processing

As an inseparable and important companion to the approach for selection and development of alternative treatment and disposal technologies for aluminum-based SNF, the question of how best to utilize the existing SRS processing capability deserves careful consideration.

The Team was convened to identify and evaluate alternatives to processing in the SRS canyons, and that has been the clear focus of this work. Nonetheless, the SRS processing capability is in place and is currently planned to be used until 2002. While the mission for the canyons is being finalized, several SNF types have already been designated for processing during this time frame. These fuel types include the Mk-16 and -22 SNF and the Mk-31 targets, 81 of 143 canisters of degraded TRR SNF, and one canister containing EBR-II SNF.¹⁷ The processing of these fuels over the next few years will significantly alleviate the SNF management challenge at the SRS. At the same time, it is likely that modest additions to the list of SNF approved for processing may greatly improve the cost-effectiveness of the overall research reactor SNF treatment and disposal campaign. In short, selective processing should be a key element in the composite approach.

Table 5.2-1 is a list of SNF candidates that should be considered for selective processing during the period that the SRS processing facilities are operating. The SNF types are arranged in priority order. Leading the list are the aluminum-based metallic uranium fuels. This fuel type is most vulnerable to corrosion in wet storage, and the attendant release of fission products to pool water is a potentially significant health and safety concern. DOE has already elected to process a portion of this SNF. If the remainder were to be processed, all concerns regarding degradation and fission product release from the fuel would be eliminated within the shortest possible time frame. If this SNF is not processed, a more expensive alternative, such as providing hot vacuum drying facilities at SRS, or shipping the SNF to Hanford to use their hot vacuum drying facilities, may be required.

Oxide target SNF is the next category on the list. This is a small quantity of SNF that, if processed, would eliminate concerns over the placement of powdered (particulate) SNF in the repository. Likewise, it is recommended that the very small quantities of failed and/or sectioned SNF be considered for processing. If the oxide target materials and the failed/sectioned SNF were processed, the remaining SNF requiring treatment would be limited primarily to aluminum-clad, aluminum-uranium alloy fuel types.

Table 5.2-2 identifies FRR SNF that may be received in the future under the FRR Environmental Impact Statement (EIS) and may also warrant additional consideration. The SNF in this category consists primarily of target materials that, in the past, were received in particulate form. Currently, much of this material is in liquid form. This FRR SNF is of concern from the standpoint of both storage and repository disposal requirements.

Table 5.2-1 Aluminum-Clad SNF Processing Candidates¹

Fuel Type	Reason for Processing	Alternatives	Number of Items	Mass (MTHM)	Volume ³ (m ³)	Dissolving Time ² (Years)
Metallic Uranium Fuels:						
Taiwan Research Reactor	Poor cladding condition and historical failures. Uranium metal will react quickly with basin water if canister leakage occurs, with release of fission products to the basin and the potential for gross canister failure. Not suitable for direct disposal without treatment.	Ship to Hanford for hot vacuum dry treatment. Store in Canister Storage Building being constructed at Hanford.	62 cans	16.11	1.47	0.4 X-H
Experimental Breeder Reactor-II	Fuel is declad. Uranium metal will react quickly with basin water if canister leakage occurs, with release of fission products to the basin and the potential for gross canister failure. Uranium metal fuels may not be suitable for repository emplacement because uranium metal is not chemically stable (forms oxides or hydrides). Not suitable for direct disposal without treatment.		59 cans	16.58	3.00	0.8 X-H
Sodium Reactor Experiment	Fuel is declad. The uranium-thorium metal will react quickly with basin water if canister leakage occurs, with release of fission products to basin. Not suitable for direct disposal without treatment.		36 cans	2.127	0.02	0.2
Oxide Target Materials (particulate residues from medical isotope production):						
Sterling Forest Oxide	Particulate materials would readily disperse in basin water if canister leakage occurs, with release of fission products to the basin. Repository emplacement of particulate material is not allowed.	Ship to INEL for stabilization using the Electrometallurgical Treatment option.	878 cans	0.102	3.7	0.6
Failed and Sectioned Fuels:						
Oak Ridge Reactor (canned silicide, oxide, aluminide pieces)	Placing cans of failed and sectioned fuels into existing storage basins presents potential environmental, safety, and health concerns. Future canning and characterization would result in additional personnel exposure and expense. The quantities of sectioned fuels are very small.	Store at SRS in a suitable facility for dry storage pending resolution of disposition path.	9 cans	0.0195	0.1	0.3
High Flux Isotope Reactor (canned oxide fuel pieces)			1 can	0.0000126	0.011	
Tower Shielding Reactor (canned aluminide fuel)			2 cans	0.00025	0.022	
Tower Shielding Reactor (aluminide fuel)			1 element	0.0092	0.1	
<p>Notes: 1. The total volume of 8.42 m³ listed in this table represents 3.3 percent of the 255 m³ of DOE-owned aluminum-based SNF expected to be in inventory by 2035. The total mass of 34.9 MTHM represents 56.3 percent of the 62 MTHM expected to be in inventory by 2035.</p> <p>2. The SNF materials can be processed in either the F or H canyon unless an "X-H" is present, indicating that the SNF materials can not be processed in H-canyon because it is not configured to process depleted or natural uranium fuel at any practical throughput.</p>						

Table 5.2-2 FRR SNF Requiring Additional Consideration

Fuel Type	Reason for Processing	Alternatives	Mass (MTHM)	Volume (m³)	Dissolving Time² (Years)
Target materials in powdered form to be received under FRR EIS (Canada, Belgium, Argentina, and Indonesia)	Similar materials have previously been received in particulate form. ¹ Particulate materials would readily disperse in basin water if canister leakage occurs, with release of fission products to the basin. Repository emplacement of particulate material is not allowed.	Explore receiving the material in a different form (that is suitable for direct disposal) from the reactor operators. Store in Canister Storage Building being constructed at Hanford.	0.56	6.5	3.5 ²

- Notes:
1. It has been assumed that this material will be packaged, and in a form similar to that of the Sterling Forest Oxide SNF.
 2. Much of this material is currently in a liquid form and has been shipped in the past in powdered form. It may be possible to dilute and solidify the fuel in a form suitable for disposal at the foreign reactor site prior to shipment to the U.S. This option will be evaluated during the R&D effort recommended to be conducted by SRS.

■ 5.3 Implementation

DOE's objective – to have in place by the year 2000 a committed, high confidence path forward for the treatment and disposal of the research reactor SNF – is a very demanding one. The technical strategy recommended here can achieve that objective, but (in the Team's view) only if the work is managed as a fast track, high priority program.

Over the next four years, the program should involve parallel tracks, including further evaluation of selected alternatives, conceptual design, policy determinations, interactions with regulators, funding actions, review and decisions on technical matters, and the like. Also included will be some design/build activities, and some SNF processing work in selected areas. The following is a list of key activities which require near-term attention.

Technical and Economic Evaluations

SNF Handling, Transfer, and Pre-Treatment Storage Options

Initiate more complete technical and economic evaluation of the most promising options and their implications, leading to a formal recommendation to DOE management with respect to the SNF transfer facility. This must include consideration of continued use, with modification as necessary, of the existing wet storage facilities at SRS, and an economic assessment of the cost-benefit of an early start of this facility, with treatment and storage modules added later.

Definition and Scheduling of FRR Receipts

Develop a more complete planning basis as to the timing, types, quantities, conditions, operating histories, and packaging of these SNF materials.

Selective Processing

Initiate further refinement, and technical and economic evaluation of the candidates for processing outlined in Section 5.2, leading to a formal recommendation to DOE management.

Point-of-Origin Packaging

The economics and efficiency of the treatment alternatives under consideration, and particularly the Direct Co-Disposal approach, may be substantially improved by characterization and pre-packaging at the point-of-origin. This should be explored early in the program.

Interim Post-Treatment Storage

All options under consideration involve some interim period of dry storage (probably several decades, or longer). An evaluation of available systems, costs, siting, etc. is required as a prerequisite to initiating conceptual design of needed facilities, to determine an initial set of interface requirements for the treatment processes. Options include the use of the DWPF glass waste storage building, or the construction of a new facility.

Safety Analyses

Repository Post-Closure Criticality Safety

As discussed in Section 3.1, a review of existing studies and additional work, as needed, should be conducted on a priority basis to establish the technical foundation for waste form requirements, vis-a-vis disposal of HEU.

Policy Actions

DOE Endorsement of Selective Processing

Based on recommendations in this report (and as refined by the evaluation described above), DOE management should endorse and secure funding for processing of identified SNF.

Regulatory Actions

Office of Civilian Radioactive Waste Management

Gain formal agreement within DOE on waste form requirements for research reactor SNF.

Nuclear Regulatory Commission

Secure NRC endorsement of proposed DOE requirements for the disposal of research reactor SNF. Particularly sensitive issues include waste form qualification (addressing packaging requirements, spacing, material form and stability, pyrophoricity, etc.) and criticality control. Continue the interaction with the NRC to secure their acceptance of DOE's technical work in demonstration of satisfaction of technical requirements.

Funding Actions

Overall Program

Put in place funding as required to establish and support the organization(s) needed for the programmatic, technical, and regulatory actions described herein.

SNF Transfer Facility

Put in place funding to support engineering, design, and construction (in successive stages) to achieve the mortgage reduction afforded by early start of the transfer facility.

Conceptual Design

Development of Direct Co-Disposal and Press (or Melt) and Dilute

Further develop the recommended options, through the conceptual design phase, to the degree necessary to support a definitive path forward decision by DOE.

SNF Transfer Facility

Proceed with conceptual design of the transfer facility, per this report and subsequent evaluations.

Program Management

Program Plan

A program management plan, identifying decision dates, schedules, budget requirements, interfaces among responsible agencies (DOE, OCRWM, NRC, stakeholders, etc.), and defining overall roles and responsibilities, is needed early on.

The overall set of tasks, and their relative timing, is shown schematically in Figure 5.3-1.

As noted above, the Team considers these actions to be complex and interrelated tasks requiring management by a single Program Management organization. The structure and staffing of this organization is likely to be a crucial element in the success of this effort.

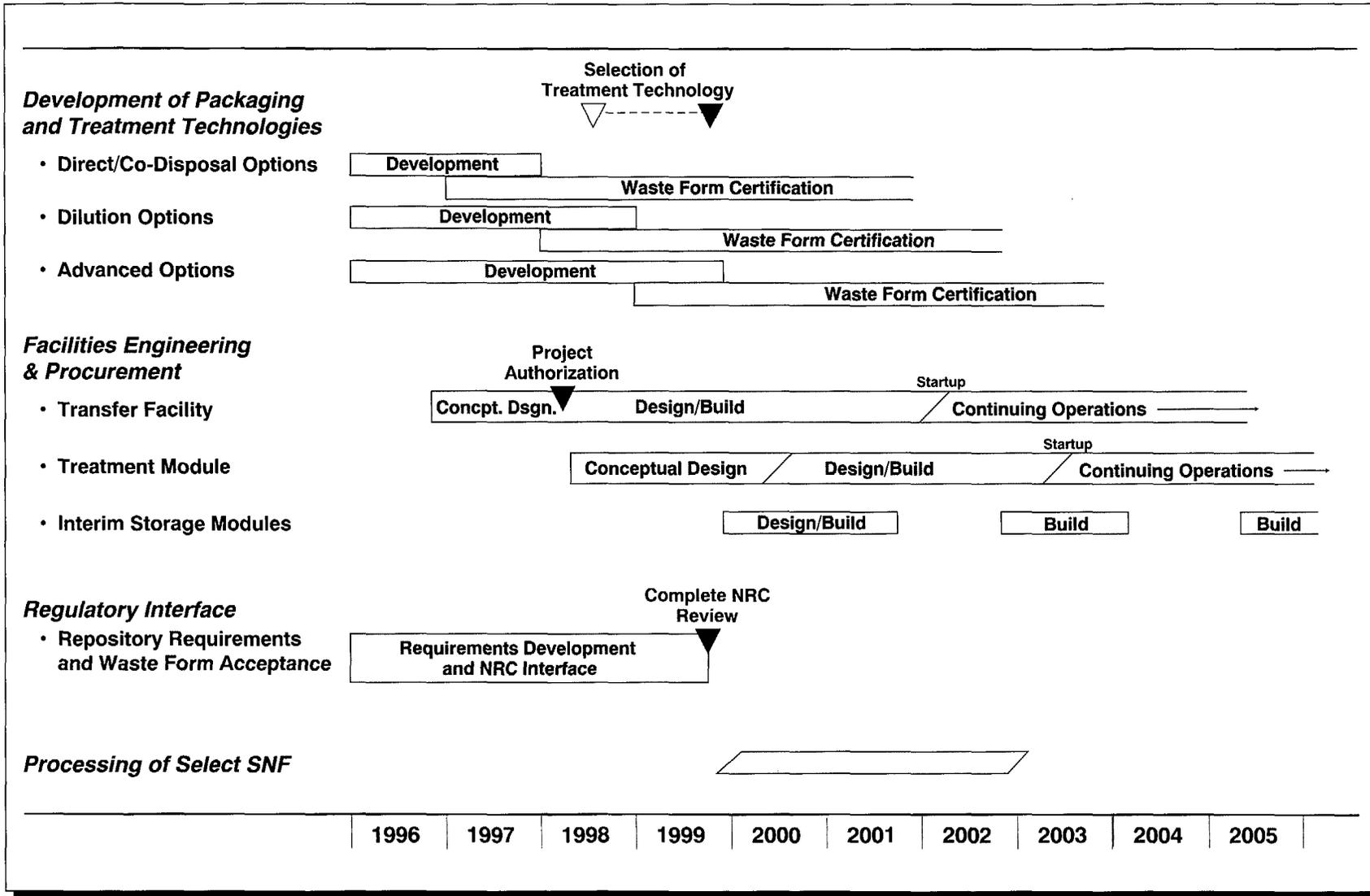


Figure 5.3-1 Implementation Plan Timeline

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6. Findings and Recommended Actions

6.1 Findings

Based on its evaluation and comparison of alternatives, the Team reached the following conclusions:

- Numerous viable options exist for the safe, cost-effective treatment, packaging, storage, and disposal of research reactor SNF. DOE's timetable can be met.
- A determination of waste form acceptance is essential for all of the options considered. All of the treatment options are capable of producing technically satisfactory waste forms, but some of the waste forms may be more difficult to qualify than others (in that they are significantly different than commercial SNF waste forms). Just as important, the process of gaining high confidence of waste form acceptance, in the time frame required, may be the most challenging aspect of the work.
- As a packaging strategy, co-disposing any of the waste forms with packaged vitrified HLW appears technically sound and cost-effective.
- For those few fuel types that present technical challenges for treatment by the candidate technologies, processing is an attractive option and could be accomplished while SRS processing capability remains available.
- For all technologies considered, a new facility for receipt, handling, and packaging of fuel appears cost beneficial in that it would permit early closure of the existing SRS wet basins.

6.2 Recommendations

Based on these findings, the Team recommends the following course of action:

- DOE should proceed with the parallel development of at least two treatment technologies. The recommended choices are:
 - **Direct Co-Disposal** as the primary approach.
 - A dilution option, **Press and Dilute** or **Melt and Dilute** (as selected by further evaluation), as the backup.
- DOE should utilize the existing SRS processing capability for those few fuel types for which processing clearly is the most cost effective and timely treatment method.

- DOE should retain an advanced technology option, as a secondary and diverse backup, should one be needed. Electrometallurgical Treatment was evaluated as a potentially strong technical candidate. Its development is well along and will continue without additional funding while the direct disposal and the dilution technologies are being developed.

At this stage, pursuing development of multiple options provides high confidence that a cost-effective treatment method, yielding a satisfactory waste form, will be available in the time frame required. For the research reactor fuel, the final approach may involve one or more of these methods, in concert with selective processing.

- DOE should begin immediately to work with the NRC and other regulatory authorities, to evaluate and reach agreement on SNF disposal requirements, and particularly on HEU waste forms.
- DOE should move ahead with the planning, funding, and conceptual design for an SNF transfer facility. It should be designed to accommodate a treatment module based on technology selection(s) made later.
- To achieve the schedule desired by DOE, a strong, focused and results-oriented effort is required. It is recommended that a project approach be employed.

■ 6.3 Conclusion

In November 1995, DOE charged the Research Reactor Spent Nuclear Fuel Task Team with developing a recommended strategy for the treatment, packaging, and disposal of the aluminum-based SNF in hand or expected to be received at the Savannah River Site. The Team has developed such a technical strategy and presents it in this report.

The recommendations herein do not propose a final, definitive treatment and disposal method, but rather a logical course of action for DOE to develop such a path, building on the information in hand and recognizing the uncertainties and technical challenges that remain. The recommended strategy involves parallel conceptual development of several options, along with supporting regulatory activities and design and construction of facilities which would be common for all approaches. Also included in this strategy is the processing, using existing SRS canyon processing capability, of selected fuel types for which that approach is clearly optimal.

The Team is confident that DOE's schedule and performance objectives can be met, provided that this work proceeds on a fast track, high priority basis. We appreciate this opportunity to assist DOE in this important effort.

7. References

- ¹ *Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel*, DOE/EIS-0218F, February 1996.
- ² *Waste Acceptance System Requirements Document*, Revision 2 DCN 1, DOE Office of Civilian Radioactive Waste Management, May 1996.
- ³ *Preliminary Requirements for the Disposition of DOE Spent Nuclear Fuel*, A00000000-00811-1708-00006, Revision 0, TRW Environmental Safety Systems, December 1995.
- ⁴ *Performance Assessment of the Direct Disposal in Unsaturated Tuff of Spent Nuclear Fuel and High-Level Waste Owned by the U.S. Department of Energy*, SAND94-2563, Sandia National Laboratories, March 1995, p. ES-6.
- ⁵ 10 CFR Part 60.131(b)(7).
- ⁶ "Comments on Proposed Changes to 10 CFR Part 60 Related to Design Basis Events," *Federal Register* Vol. 60, No. 55 (60 FR 15180), Letter from R. Milner (RW-30) to U.S. Nuclear Regulatory Commission Secretary, March 22, 1995.
- ⁷ U.S. Department of Energy Annotated Outline for Topical Report, "Disposal Criticality Analysis," Letter from Stephan J. Brocoum (YMSCO) to Michael J. Bell (U.S. Nuclear Regulatory Commission), April 12, 1996.
- ⁸ *Assessment of Technical Issues and Cost for Direct Disposal of Foreign Research Reactor Fuel*, A00000000-01717-5705-00010, Revision 0, TRW Environmental Safety Systems, December 1995.
- ⁹ DOE-Owned Spent Nuclear Fuel Technology Integration Plan, DOE/SNF/PP-002, Revision 1, May 1996.
- ¹⁰ *An Assessment of Continued R&D into an Electrometallurgical Approach for Treating DOE Spent Nuclear Fuel*, National Research Council, Committee on Electrometallurgical Techniques for DOE Spent Fuel Treatment, Washington, D.C., 1995, p. 19.
- ¹¹ *The New Rational Manager*, Charles H. Kepner and Benjamin B. Tregoe, Princeton, NJ, 1981, p. 41.
- ¹² *Waste Acceptance System Requirements Document*, Revision 2 DCN 1, DOE Office of Civilian Radioactive Waste Management, May 1996.
- ¹³ *Preliminary Requirements for the Disposition of DOE Spent Nuclear Fuel*, A00000000-00811-1708-00006, Revision 0, TRW Environmental Safety Systems, December 1995.
- ¹⁴ Andrews, R. W., et al., "Total System Performance Assessment - 1995: An Evaluation of the Potential Yucca Mountain Repository," B00000000-01717-2200-00136, Rev. 01, Civilian Radioactive Waste Management System, Management and Operating Contractor, Las Vegas, NV, November 1995.

- 15 "Waste Management Facilities Cost Information for Spent Nuclear Fuel," EGG-WM-10670, and "Waste Management Facilities Cost Information Estimating Data for Spent Nuclear Fuel," EGG-WM-10708, March 1993.
- 16 *The New Rational Manager*, Charles H. Kepner and Benjamin B. Tregoe, Princeton, NJ, 1981, p. 41.
- 17 "Record of Decision for the Interim Management of Nuclear Materials at the Savannah River Site," 60 FR 65300, December 19, 1995, and the "Supplemental Record of Decision," 61 FR 6633, February 21, 1996.

■ **Curricula Vitae**

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Professor Ballinger has been called upon to apply his expertise in resolving a variety of technical issues facing the Department of Energy (DOE). Most recently, he participated as a member of the Independent Technical Assessment Team for Dry Storage of N Reactor Fuel at Hanford. This team established the technical feasibility and developed a conceptual engineering approach for packaging, transport, stabilization and dry storage of spent nuclear fuel at the Hanford K Basins; its recommendations were adopted by DOE and are being implemented.

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Mr. Benton served with the U.S. Navy for 29 years, mostly in the nuclear power program. His assignments included service on the new construction and operating crews of four nuclear powered submarines, command of groups of up to ten nuclear powered and support ships, and command of the Charleston, SC Naval Base which included a nuclear qualified shipyard. He was also in charge of the team responsible for the inspection of nuclear powered ships in the Pacific Fleet. He completed his naval service as Deputy Chief of Naval Personnel.

Mr. Benton was employed by the Babcock & Wilcox Company for 10 years. As Manager and later Vice President for Ventures and Licensing of Babcock & Wilcox International, he established and supervised companies including manufacturing plants in China, India, and Indonesia, and engineering and marketing companies in Mexico and Turkey. In 1991, Mr. Benton joined the B&W Fuel Company (now Framatome Cogema Fuels), then a subsidiary of

the Babcock & Wilcox Company, as a Manager of Waste Package Development for the OCRWM M&O contractor. The group currently consists of 24 scientists and engineers, and is developing the technical basis and designing the containers for commercial spent nuclear fuel and defense high-level waste, as well as the other engineered barriers in the potential repository.

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Mr. Bradley has more than 25 years of experience in the commercial nuclear industry and is currently a senior project engineer at Duke Engineering and Services Inc. Mr Bradley has architectural and engineering experience with projects related to spent fuel storage, and is a member of the American Nuclear Society Committee 57.9 (spent fuel storage) and the American Society for Testing Materials Committee E10.11.06 (spent fuel storage and transportation).

Mr. Bradley has provided support in the area of spent fuel storage to commercial utilities and the Department of Energy. He participated as a member of the Independent Technical Assessment Team for Dry Storage of N Reactor Fuel at Hanford, evaluating the feasibility of stabilization and dry storage of N Reactor spent nuclear fuel at the Hanford K Basins. In addition, he is a member of the Technical Assistance Group to provide support for the design, licensing, construction, and operation of a dry spent fuel storage facility for N Reactor.

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An attorney and engineer, Mr. Burns has over 20 years experience in the nuclear field, from nuclear weapons in the Army to nuclear licensing at Westinghouse. As nuclear plant licensing manager in the 1980's, he managed NRC, ACRS and ASLB licensing actions associated with the startup of the Diablo Canyon, Comanche Peak, and Vogtle nuclear plants in the U.S. and the Krsko (Yugoslavia), Maanshan (Taiwan), and Korea 5-8 nuclear plants overseas. Since 1988, he has developed environmental and regulatory strategies for advanced plant designs including the Westinghouse Advanced Passive (AP-600) nuclear plant, a hardened 10 Mwe plant for the Air Force, and a new heavy water reactor for the Department of Energy. Mr. Burns has worked as a senior technical and regulatory advisor in DOE's Office of Spent Fuel Management since its inception in 1993.

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Mr. Conatser has more than 25 years of engineering and project management experience in the chemical and nuclear industries. He worked at Dupont in the Petrochemicals Business Units for 13 years, carrying out a number of engineering and operations assignments. Mr. Conatser began work with the Westinghouse Savannah River Company in 1982. Since that time, he has functioned in a variety of professional and management assignments, including Facility Manager for a high pressure high temperature oxidation operation, and System Engineer for equipment and materials design. He was the Project Manager for the facility modifications completed in preparation for the 1991 restart of the Savannah River Site (SRS) K reactor. Other assignments while at SRS have included Chief Engineer for the Reactor Materials Fabrication and the Mechanical Systems Engineering organizations, and Special Equipment and Test Facility Design Engineer.

Mr. Conatser is currently the Westinghouse Savannah River Company Spent Nuclear Fuels Program Manager. He has been in this assignment for the past 4 years, and is responsible to the Department of Energy (DOE) for development of the Program Plan to disposition the spent nuclear fuel (SNF) received at the SRS from off-site DOE research reactors, and integration of this mission with the ongoing stabilization of SRS irradiated SNF.

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Dr. Detrick has more than 29 years of experience in the design, analysis, manufacture and construction, testing, and operation of naval nuclear propulsion plants. He also has in-depth experience in emergency planning, radiological and hazardous waste management, and environmental evaluation for the Naval Nuclear Propulsion Program and in the development of models, methods, and computer programs for neutron transport, kinetics, and shielding calculations for nuclear reactors.

Dr. Detrick has been employed by the Westinghouse Electric Corporation at Bettis Laboratory since 1967. He is currently Manager, Environmental Affairs Activity at Bettis, reporting to the Laboratory General Manager. He has qualified as a Test Physicist, shipyard Joint Test Group member and Joint Refueling Group member, Engineering Officer of the Watch, and Prototype Plant Manager within the Naval Nuclear Propulsion Program. He has been associated at one time or another with the design, operation, or analysis of every naval nuclear core design under Bettis cognizance. He has served on the Bettis Reactor Operating Safety Committee and is currently serving for the second time on the Bettis Advisory Safety Committee. He is also a member of the Naval Reactors Senior Panel for Human Reliability Assessment.

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U.S. Navy Nuclear Power Training Program

Mr. DeVine has more than 30 years of nuclear power experience including service with the U.S. Navy as a commissioned officer in the nuclear submarine force. Mr. DeVine also is experienced in the recovery of degraded nuclear fuel, including work as Recovery Engineering Manager at Three Mile Island Unit 2 following the March 1979 nuclear accident.

Mr. DeVine was employed with General Public Utilities Corporation (GPU) for 22 years, most recently as Vice President and Director, Technical Functions, responsible for engineering support of the corporation's nuclear stations. He was a member of the GPU Nuclear Board of Directors, the Project Management Board of the Advanced Reactor Corporation, and the Executive Board of the Edison Electric Institute Utility Waste Management Group.

Mr. DeVine is co-founder of Polestar Applied Technology, Inc., a management and engineering services firm. He is currently providing technical support to the Department of Energy (DOE) in the evaluation of spent nuclear fuel options at the Savannah River Site. He also functions as an Independent Technical Expert providing review and assistance to the DOE (EM-63) for the deactivation of the PUREX/UO₃ facilities at the Hanford Site, and ROVER fuel at the Idaho National Engineering Laboratory. Previously, he served as Chairman of the Independent Technical Assessment Team for Dry Storage of N Reactor Fuel at Hanford. This team established the technical feasibility and developed a conceptual engineering approach for packaging, transport, stabilization and dry storage of the spent nuclear fuel at the Hanford K Basins.

■ ■ ■ **DENZEL L. FILLMORE**

Ph.D. Inorganic Chemistry, Brigham Young University
B.S. Inorganic Chemistry, Brigham Young University

Dr. Fillmore has 25 years experience in a diversity of nuclear industry positions. He has extensive knowledge in nuclear fuel design, characteristics, material properties, potentially hazardous material, and the Department of Energy (DOE) fuel inventory.

Dr. Fillmore began his career with the Bettis Atomic Power Laboratory at the Naval Reactor Facility by qualifying on a nuclear submarine prototype. He then qualified Westinghouse and Naval personnel to operate Naval Nuclear Reactors. For the last 8 years, Dr. Fillmore has worked at the Idaho National Engineering Laboratory (INEL) in the Spent Nuclear Fuel Programs. He aided in the establishment of both the INEL and National SNF programs. He participated in the development of the DOE SNF Technology Integration Plan, and developed the SNF inventory and characteristics information used in the preparation of the INEL Environmental

Impact Statement. Presently, Dr. Fillmore is supporting the DOE National and INEL SNF programs in the areas of SNF characterization, data collection and analysis, and program development.

■ ■ ■ **JOSEPH F. KRUPA**

M.E. Chemical Engineering, University of Idaho

M.S. Chemistry, University of California Berkeley (AEC Fellowship in Nuclear Science and Engineering)

B.S. Chemistry, U.S. Air Force Academy

Mr. Krupa has over 20 years experience in the nuclear field. He began his career as a Nuclear Research Officer in the U.S. Air Force. He then spent 10 years at the Idaho Chemical Processing Plant studying actinide removal from spent fuel waste. He developed Flourinel Dissolution Process reagent addition computer programs for which he was awarded the George Westinghouse bronze award in 1985, and was a key player in the successful modification and implementation of the Flourinel Process for Naval Fuel dissolution.

Mr. Krupa spent 5 years as a Nuclear Engineer for the Department of Energy (DOE) Savannah River Operations Office. During his tenure, he acted as DOE Nuclear Materials Manager responsible for coordination and review of technical planning studies on nuclear materials disposition, transportation, and capital asset management. Most recently, Mr. Krupa has functioned as a Principal Engineer for Westinghouse Savannah River Company, publishing a study of aluminum-clad spent fuel options to support DOE Environmental Impact Statement Records of Decision. He has co-authored studies of life-cycle costs for spent fuel disposition with criticality prevention, Savannah River Site (SRS) spent fuel storage, SRS plutonium discard limit implementation, SRS nuclear materials disposition, and complex-wide nuclear material disposition issues.

■ ■ ■ **HENRY H. LOO**

B.S. Chemical Engineering, University of California Berkeley

Mr. Loo has more than 20 years experience in areas such as performance assessment, and waste form criteria and requirements for disposal of spent nuclear fuel and high-level waste. His background includes process design, procurement, and installation of a radioactive fluid bed waste calcination and a fuel reprocessing process; and process design of a vitrification facility for radioactive waste.

Mr. Loo has functioned as a contractor at the Idaho National Engineering Laboratory (INEL) for 15 years. In this capacity, Mr. Loo has managed work assignments aimed at determining the performance of various spent fuel and waste forms destined for permanent disposal in geologic repositories. He has also been involved in the development of the waste form criteria and requirements document for repository disposal. In addition, he spent 3 years as a process

engineer for the Defense Waste Processing Facility. Mr. Loo is currently employed at the INEL with Lockheed-Martin Idaho Technologies Company, working on the Department of Energy Spent Nuclear Fuel programs.

■ ■ ■ **PHYLLIS M. LOVETT**

M.S. Nuclear Engineering, Massachusetts Institute of Technology

B.S. Nuclear Engineering, Georgia Institute of Technology

Ms. Lovett has over 10 years experience in the nuclear industry, in both the commercial and government arenas. Ms. Lovett earned her Master of Science degree from the Massachusetts Institute of Technology (MIT) in the area of experimental thermal hydraulics, and was selected as an Institute of Nuclear Power Operations fellow.

Ms. Lovett began her career with Virginia Power, as a licensing engineer working with the Nuclear Regulatory Commission on projects related to the Surry Power Station independent spent fuel storage installation, power plant physical protection, emergency planning, and decommissioning. She then became Shift Technical Advisor working on shift as part of the operations team for the Surry nuclear unit.

Ms. Lovett has spent the last 5 years with TRW Environmental Safety Systems, Inc., the Management and Operating contractor for the Department of Energy (DOE) Office of Civilian Radioactive Waste Management (OCRWM). Ms. Lovett is currently the lead technical contractor on DOE spent nuclear fuel and other high-level waste for the OCRWM. She has authored the baseline requirements for both waste acceptance and transportation by the U.S. high-level waste program, as well as other waste form criteria documents.

■ ■ ■ **JOHN M. McCONAGHY, JR.**

M.E. Virginia Polytechnic Institute and State University

B.S. United States Military Academy

Mr. McConaghy has more than 16 years of experience in the nuclear power industry. As an employee of Duke Power Company, he was involved in structural design of each of their nuclear generating stations. Mr. McConaghy also developed structural design requirements for the Advanced Light Water Reactor program of the Electric Power Research Institute.

As an employee of Duke Engineering and Services, a subsidiary of Duke Power Company, Mr. McConaghy supervised a group responsible for conceptual design of a Department of Energy interim storage facility for commercial spent nuclear fuel. This effort included demonstration of concept feasibility, development of facility criteria and concepts, and calculation of facility life cycle cost estimates.

■ ■ ■ **THOMAS P. McLAUGHLIN**

Ph.D. Nuclear Engineering, University of Arizona

Mr. McLaughlin has over 25 years experience in nuclear criticality safety, reactor safety research, reactor design, and the performance of critical experiments. He consults to the Department of Energy (DOE), DOE contractors, and private sector firms on nuclear criticality safety and related matters. He is chairman of the American Nuclear Society Standards Subcommittee 8, which is responsible for the development of all U.S. consensus standards in criticality safety.

Mr. McLaughlin has worked at Los Alamos National Laboratory (LANL) since 1972, and is currently the Group Leader for Nuclear Criticality Safety. He is responsible for criticality safety of all weapons and nonweapons activities involving fissile materials within LANL and throughout the Department of Defense/military complex. He also designs and instructs courses in criticality safety.

■ ■ ■ **J. RICHARD MURPHY**

M.S. Mechanical Engineering, University of Texas at Austin

B.S. Mechanical Engineering, University of Texas at Austin

Mr. Murphy has been employed at the Savannah River Site (SRS) by Westinghouse Savannah River Company (WSRC), and its predecessor, I.E. Du Pont, for 14 years. Assignments have included a variety of engineering and management positions. Technical positions have been in the areas of reactor system engineering, reactor safety engineering, project management and economic development. Management positions have included reactor systems engineering and reactor field engineering teams. Other advisory and evaluation activities have included Technical Lead for the SRS Reactor Re-start Joint Test Group, and being primary participant in the development of the DOE/EM National Spent Nuclear Fuel Technology Integration Plan.

Mr. Murphy coordinates technology development activities for Spent Nuclear Fuels as part of the WSRC Spent Nuclear Fuel Program Management Group. His activities include coordinating the funding for SNF Research and Development, monitoring the progress of specific R & D tasks, integrating the R&D with the site engineering projects and interfacing with other SNF technology development programs throughout the DOE complex.

■ ■ ■ **D. KENT PARSONS**

Ph.D. Nuclear Engineering, Massachusetts Institute of Technology

M.S. Nuclear Engineering, Massachusetts Institute of Technology

B.S. Nuclear Engineering, Texas A&M University

Dr. Parsons has over 12 years of experience in computational neutronics, reactor physics, radiation shielding and nuclear criticality safety. He began his career at the Idaho National Engineering Laboratory working in the area of reactor design (e.g., the early design of the Advanced Neutron Source reactor), radiation shielding (Boron Neutron Capture Therapy at PBF), and neutronics code development (ANISN/PC).

Dr. Parsons has been with the Los Alamos National Laboratory (LANL) for the past 7 years. He has applied many different neutronics codes to the analysis of practical problems - to classified applications, to pressure vessel embrittlement analysis, and to nuclear criticality safety. Dr. Parsons currently works in the Nuclear Criticality Safety Group, where he has been responsible for criticality safety analysis at the various LANL facilities which handle and process special nuclear material. He has also been an active participant in the ongoing LANL Underground Supercriticality Review.

■ ■ ■ **ELIZABETH V. PROSSER**

B.A. James Madison University

Ms. Prosser has been employed by VPA Corporation, a management consulting and engineering services firm, since 1990. She has extensive experience in records management, technical research, and technical writing/editing.

Ms. Prosser has provided technical support to the DOE Office of Spent Fuel Management for the past three years. She has been involved in the publication of the *Environmental Assessment for the Urgent Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel*, the *Plan of Action to Resolve Spent Nuclear Fuel Vulnerabilities*, and the *DOE Spent Nuclear Fuel Interim Storage Plan*.



ELWOOD P. STROUPE

J.D. Law, University of Santa Clara

M.S. Chemical Engineering, Purdue University

B.S. Chemical Engineering, Rose Hulman Institute of Technology

Mr. Stroupe has over 30 years of management experience in spent nuclear fuel, environmental compliance, technical consulting, and nuclear power plant licensing. His extensive experience with managing technical and consulting services spans several different companies. The types of organizations he has managed include business systems, software development, engineering, nuclear licensing, financial, and project management.

Mr. Stroupe is currently the manager of the National Spent Nuclear Fuel Department at the Idaho National Engineering Laboratory which is managed by Lockheed-Martin Idaho Technologies. In this position, Mr. Stroupe routinely interfaces with numerous governmental entities, commissions, laboratories, and subcontractors.

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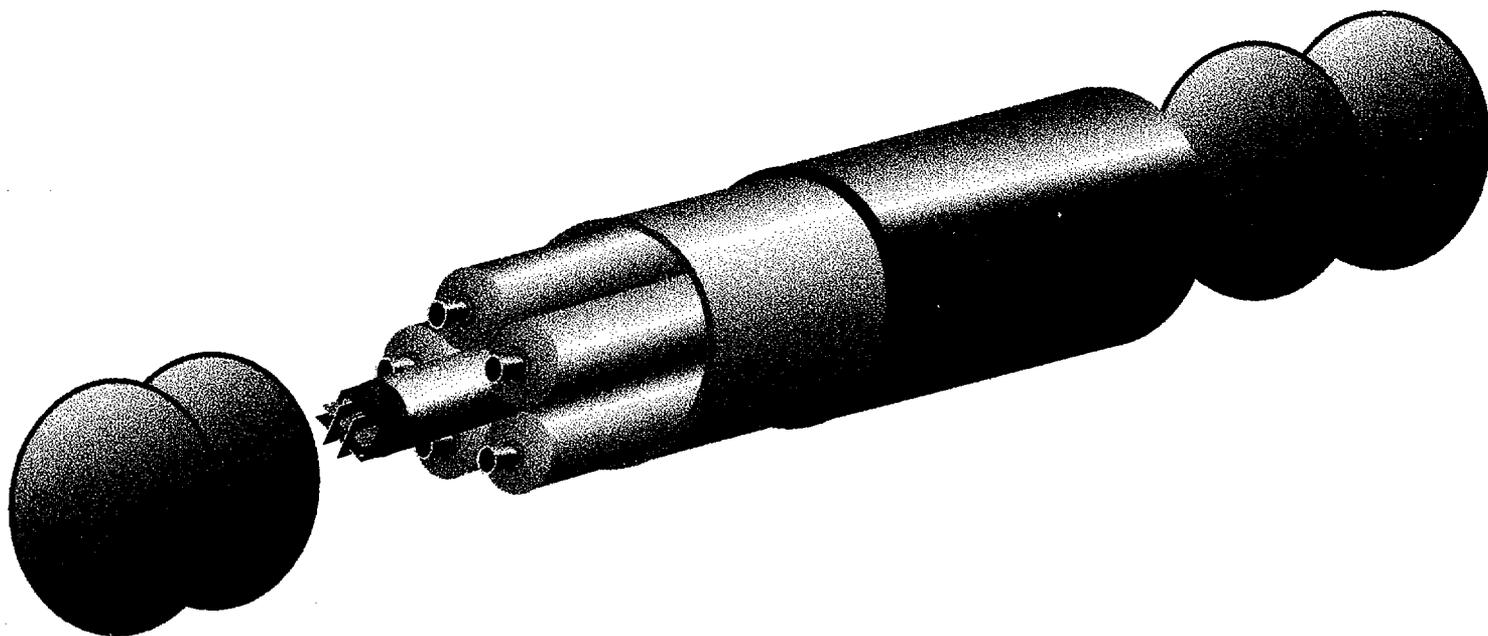
■ Acronyms

CFR	Code of Federal Regulations
DHLW	Defense high-level waste
DOE	Department of Energy
DNFSB	Defense Nuclear Facilities Safety Board
DU	depleted uranium
DWPF	Defense Waste Processing Facility
EBR-II	Experimental Breeder Reactor-II
EIS	Environmental Impact Statement
EM	The U.S. Department of Energy's Office of Environmental Management
EM-67	EM's Office of Spent Fuel Management
FRR	foreign research reactor
GMODS	Glass Material Oxidation and Dissolution System
GWSB	Glass Waste Storage Building
HEU	high-enriched uranium
HFIR	High Flux Isotope Reactor
HLW	high-level radioactive waste
INEL	Idaho National Engineering Laboratory
kg	kilogram
K-T	Kepner-Tregoe
k_{eff}	calculated effective multiplication factor
L-Basin	Savannah River Site's L-Reactoer Basin
LEU	low-enriched uranium

MPC	multi-purpose canister
MTHM	metric tons heavy metal
MTR	Materials and Test Reactor
MW	megawatt
NRC	Nuclear Regulatory Commission
NWPA	Nuclear Waste Policy Act
PUREX	Plutonium Uranium Recovery Extraction
O&M	operations and maintenance
OCRWM	Office of Civilian Radioactive Waste Management
RBOF	Savannah River Site's Receiving Basin for Offsite Fuels
RHF	Reactor a-Haut Flux
SNF	spent nuclear fuel
SRS	Savannah River Site
TRR	Taiwanese Research Reactor
U-235	uranium-235
VLEU	very low-enriched uranium

Technical Strategy for the Treatment, Packaging, and Disposal of Aluminum-Based Spent Nuclear Fuel

A Report of the Research Reactor Spent Nuclear Fuel Task Team



Prepared For
The Department of Energy
Office of Spent Fuel Management

June 1996

Appendix A
Aluminum-Based Spent Nuclear Fuel Data Summary

Spent Fuel Inventory/Database Team:

J. Richard Murphy, Westinghouse Savannah River Company

Denzel L. Fillmore, Lockheed-Martin Idaho Technologies

Larry R. Jones, Westinghouse Savannah River Company

Appendix A

Aluminum-Based Spent Nuclear Fuel Data Summary

This Appendix provides a brief overview of the DOE aluminum-based spent nuclear fuel (SNF) inventory, including existing quantities, and receipts anticipated through the year 2015.

Aluminum-Based Spent Nuclear Fuel

Quantities of SNF can be described using a variety of metrics. The most commonly used measure is Metric Tons Heavy Metal (usually uranium), abbreviated as MTHM. This unit of measure provides a convenient way to compare different fuel designs on an equivalent basis to simplify management of the material. Appendix A provides a summary of pertinent aluminum-based SNF data that describes relative quantities by type, by source (foreign or domestic), by uranium 235 (U-235) enrichment (highly enriched uranium [HEU] and low enriched uranium [LEU]), and by projected year of receipt at the Savannah River Site (SRS).

The current inventory of aluminum-based SNF at the Savannah River Site (not including SRS Production Reactor fuel and certain other failed fuels that will be chemically processed) is 54% of the projected total aluminum-based spent fuel inventory through the year 2015. As shown in Figure A-1, this current inventory of 33.5 MTHM consists mostly of Taiwanese Research Reactor fuel and EBR-II fuel, both of which contain less than 1% uranium enrichment. The remaining inventory (Figure A-1) includes both HEU and LEU from University and Domestic Research Reactors (DRR), and Foreign Research Reactor (FRR) facilities. As shown in Figure A-2, future aluminum-based spent fuel receipts are projected to consist of DRR fuels (31%), University fuels (4%), and FRR fuels (65%).

Current Inventory of Aluminum-Based Spent Nuclear Fuel at SRS

The existing inventory of aluminum-based SNF at SRS consists of SRS Production Reactor SNF and targets, University and DRR fuels, and FRR fuels. Of this total inventory, a large portion (SRS Reactor fuels and targets, as well as several Taiwanese and EBR-II elements) is scheduled to be processed in the SRS Canyon facilities and will no longer exist as SNF by the year 2000. The current inventory as discussed in this section refers to the remaining SNF that will not be processed using the SRS Canyon facilities.

The current SNF inventory contains 0.385 MTHM of HEU fuels containing 88% U-235. As shown in Figure A-3, this HEU inventory includes FRR fuels (21%), University fuels (37%), and DRR fuels (42%), based on total uranium content. The current HEU inventory total mass of 9.3 metric tons is shown in Figure A-4. Aluminum makes up 96% of the total mass of the current HEU inventory.

The current SNF inventory contains 33.1 MTHM of LEU fuels containing less than 1% U-235. As shown in Figure A-5, this LEU inventory is composed primarily of FRR (49%) and DRR fuels (51%), based on total uranium content. The current LEU inventory total mass of 57 metric tons is shown in Figure A-6. The Taiwanese fuel (FRR) and EBR-II fuel (DRR) that are not currently designated to be processed in the SRS Canyon facilities make up 98% of the current LEU inventory. Aluminum makes up 42% of the total mass of the current LEU inventory.

As shown in Figure A-7, the current inventory of HEU University fuels (0.144 MTHM) consists of fuels from Missouri University Research Reactor (63%), Rhode Island Nuclear Science Center (18%), Massachusetts Institute of Technology (13%), University of Virginia (5%), and Georgia Institute of Technology (1%). The current inventory of LEU University fuels consists of University of Michigan (96%) and Ohio State University (4%) as shown in Figure A-8.

As shown in Figure A-9, the current inventory of HEU FRR fuels (0.081 MTHM) consists of RHF (39%), R-2 (19%), Greek RR (16%), Saphir (12%), HOR (11%), and others (3%). As mentioned previously, the current inventory of LEU FRR is greater than 99% Taiwanese fuel.

Key Assumptions for Projecting Future SNF Receipts

Projections of future SNF receipts are based upon a number of key assumptions.

- For the years 1996 through 2000, fairly detailed schedules have been developed for expected shipments, by month, from University and DRR facilities within the United States to the SRS. The quantities of FRR spent fuel to be shipped during these five years are also fairly well determined (limited by ability to receive the fuel at SRS), although specific shipping locations by month or year have not been determined.
- The physical characteristics (weight, volume, uranium content, etc.) of University and DRR fuels can be applied with relative certainty to project material receipts through 2000. For FRR spent fuel, averaged values for physical characteristics must be applied to expected numbers of shipments since the specific types of FRR to be received each month and year are not yet known. It is assumed that use of these averaged characteristics will provide the best approximation to the actual FRR materials that will be received when considered over the entire five year period.
- For the years 2000 through 2015, the quantities of University and DRR fuels have been forecast based upon facility storage capacities, reactor operating and fuel discharge rates, etc. Quantities of FRR fuels have been forecast in a similar manner, assuming that these fuels will be shipped at a rate no faster than SRS can receive them. However, with the exception of HFIR cores and a few other uncommon fuel types, the specific types of fuels to be received by month and year after 2000 are not yet known. For this period,

averaged characteristics for categories of SNF were applied to forecast receipt rates for these SNF categories. It is assumed that use of these averaged characteristics will provide a valid approximation to the actual materials that will be received over the span of fifteen years.

- The majority of aluminum-based spent fuel receipts contain quantities of both HEU and LEU spent fuel. Future receipts are also expected to represent a mix of HEU and LEU materials. Projected receipts from facilities that will produce both HEU and LEU spent fuels are assumed to be in proportion to the relative quantities of these materials produced by the facility, taking into account current plans at some facilities to convert to entirely low enriched fuel. For SNF received from University and DRR facilities, the materials were assumed to be received yearly in the same proportion as the HEU/LEU proportion in their inventories.
- An additional assumption is that foreign facilities that are scheduled to convert from use of HEU fuels to LEU fuels will do so, resulting in a shift from mostly HEU FRR receipts up until the year 2000 to mostly LEU FRR receipts after 2004. This assumption is consistent with the overall portions of HEU FRR and LEU FRR fuels that are projected to make up the total FRR inventory by 2015.

Projected Total Aluminum-Based SNF Receipts

Projected aluminum-based HEU SNF yearly receipts, in kilograms uranium (MTHM times 1000), are shown in Figure A-10. Each bar in the figure indicates total uranium, as well as the portion of the material that is fissile (U-235). The higher receipt rates from 1997 through 2001 are due to the increased quantities of HEU FRR SNF expected during these years. The increase beginning in 2008 and continuing through 2012 represents the transfer of substantial quantities of aluminum-based SNF from Idaho National Engineering Laboratory to the SRS.

Projected aluminum-based LEU SNF yearly receipts, in kilograms uranium, are shown in Figure A-11. Each bar in the figure represents total uranium, as well as the portion of the material that is fissile (U-235). The higher receipt rates from 2002 through 2005 represent the expected conversion by foreign facilities from HEU fuel to LEU fuel.

Cumulative receipts, in kilograms uranium, are shown in Figures A-12 and A-13 in a similar manner for HEU SNF and LEU SNF, respectively.

Projected Aluminum-Based FRR SNF Receipts

Projected yearly receipts, in kilograms uranium, for the FRR portion of HEU SNF are shown in Figure A-14. The higher receipt rates from 1997 through 2001 are due to the increased quantities of HEU FRR SNF expected during these years. Essentially all of the FRR HEU SNF will have been received by 2008, with the exception of very small quantities

from two French research reactors and one research reactor in Belgium that are not scheduled to convert from HEU fuel to LEU fuel.

Projected yearly receipts, in kilograms uranium, for the FRR portion of LEU SNF are shown in Figure A-15. The higher receipt rates from 2002 through 2005 represent the expected conversion of foreign facilities from HEU to LEU fuels.

Cumulative FRR SNF receipts, in kilograms uranium, are shown in Figures A-16 and A-17 in a similar manner for HEU FRR SNF and LEU FRR SNF, respectively.

Figure A-1

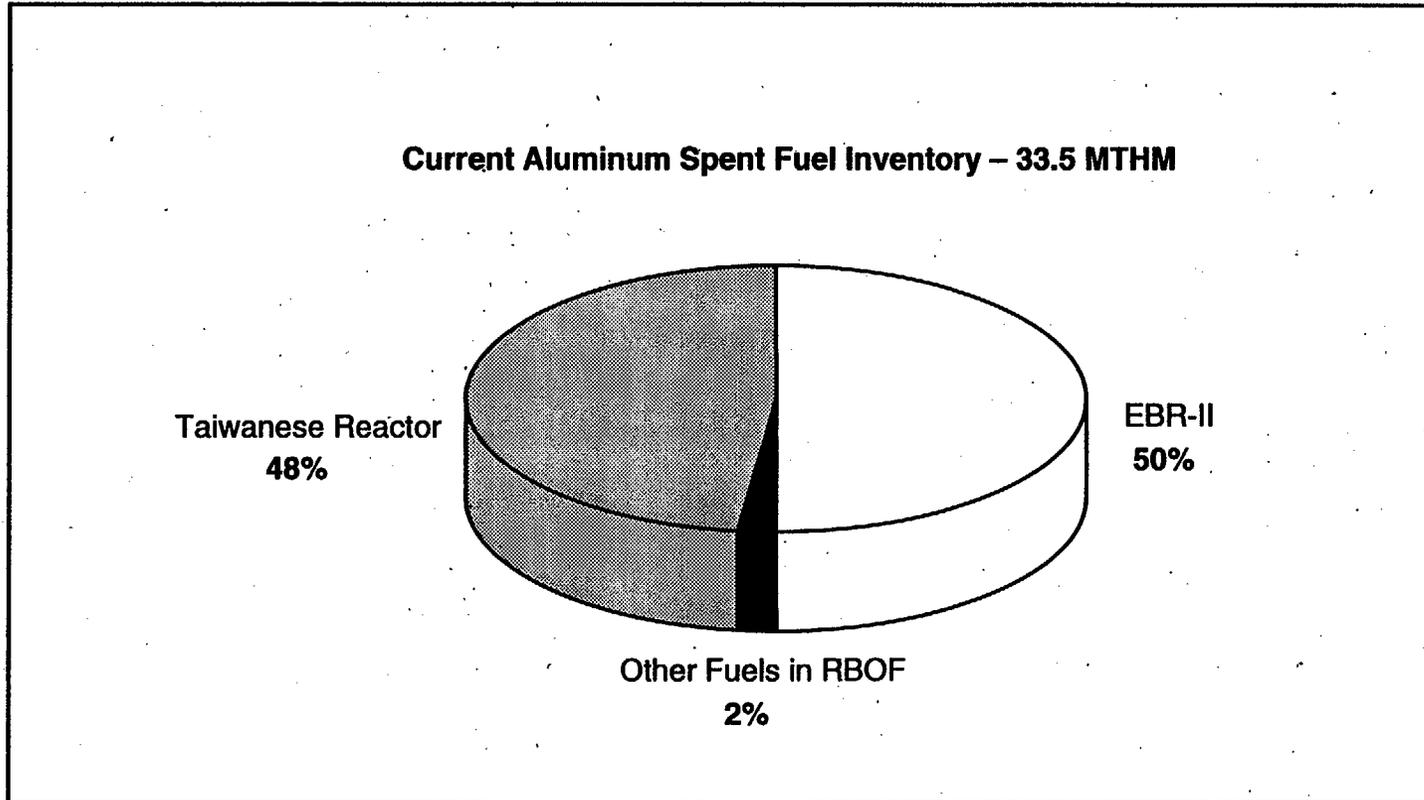


Figure A-2

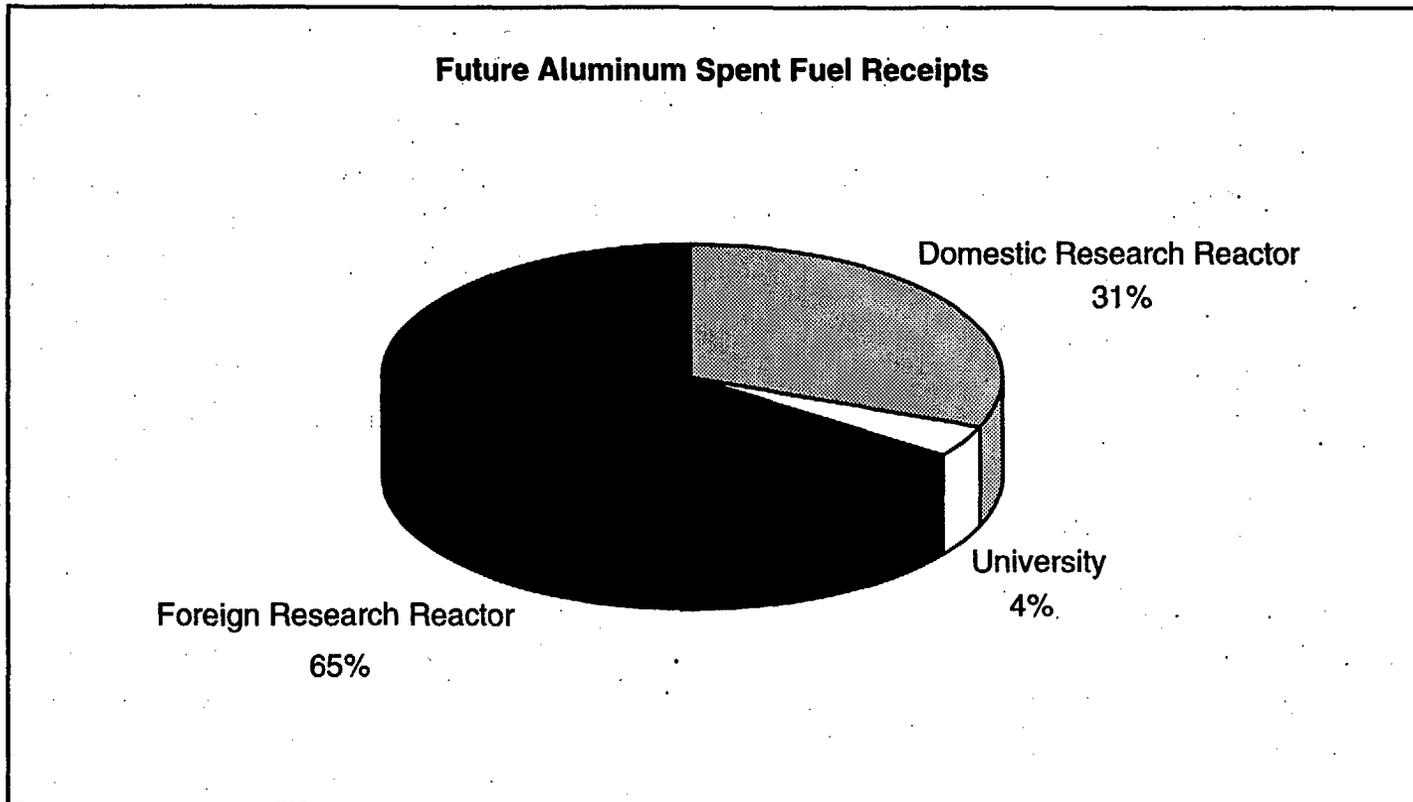


Figure A-3

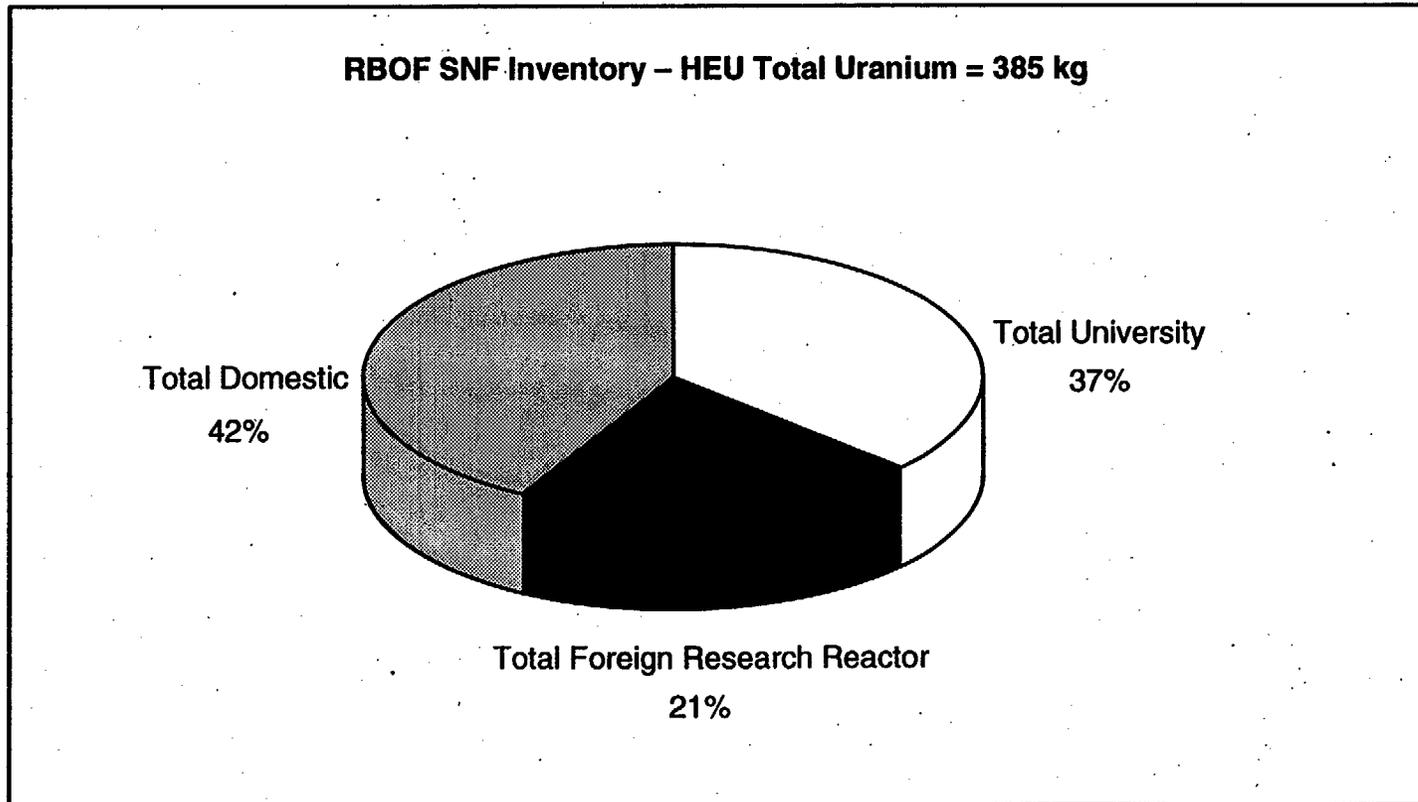


Figure A-4

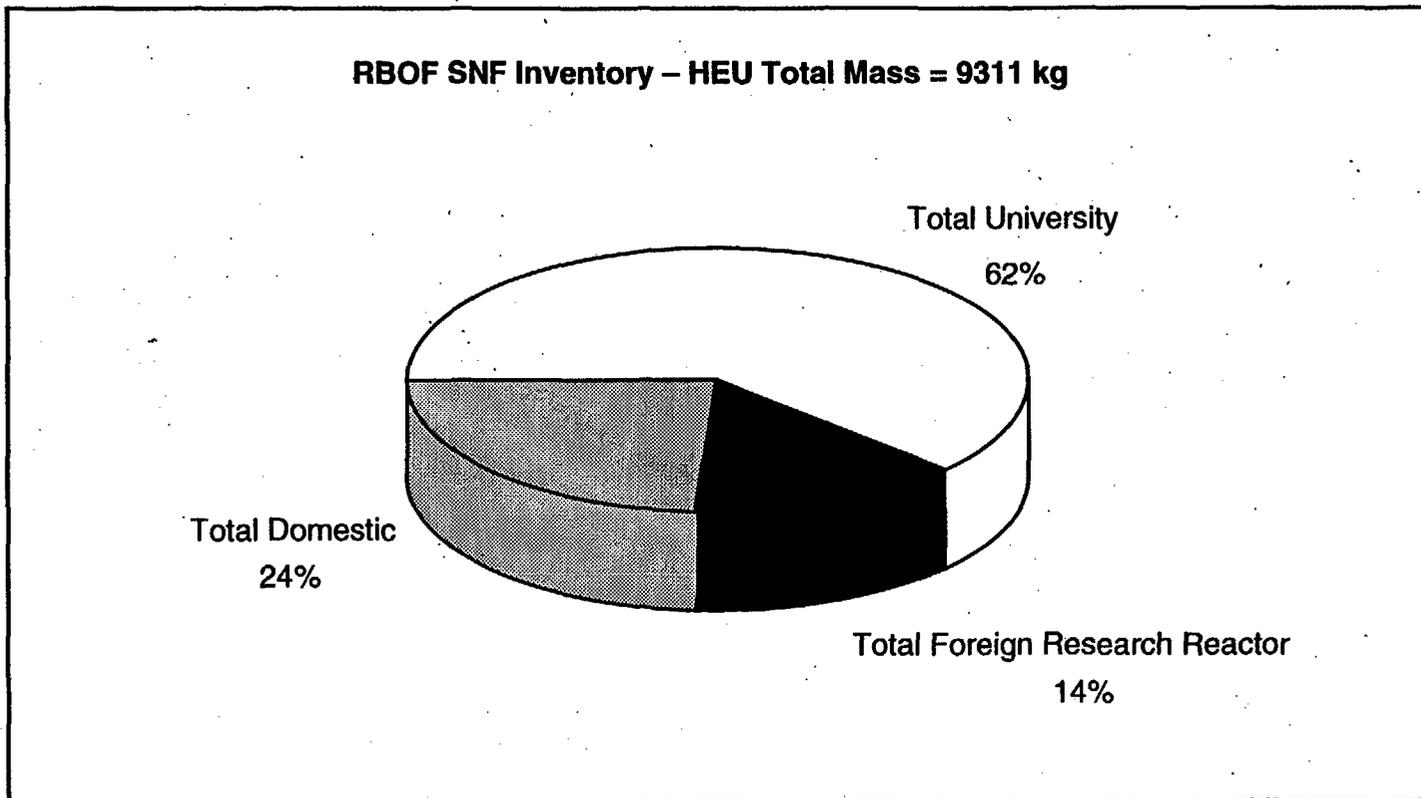


Figure A-5

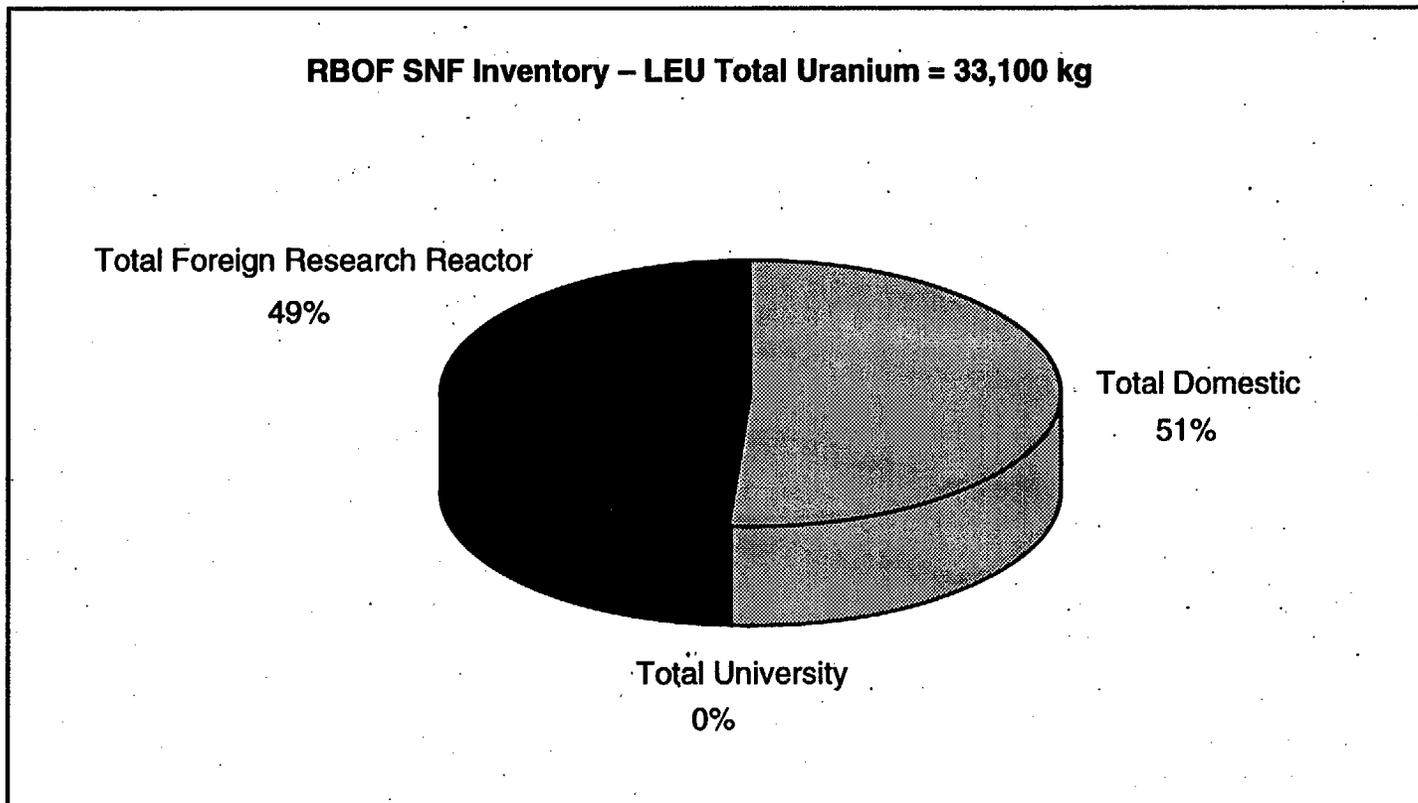


Figure A-6

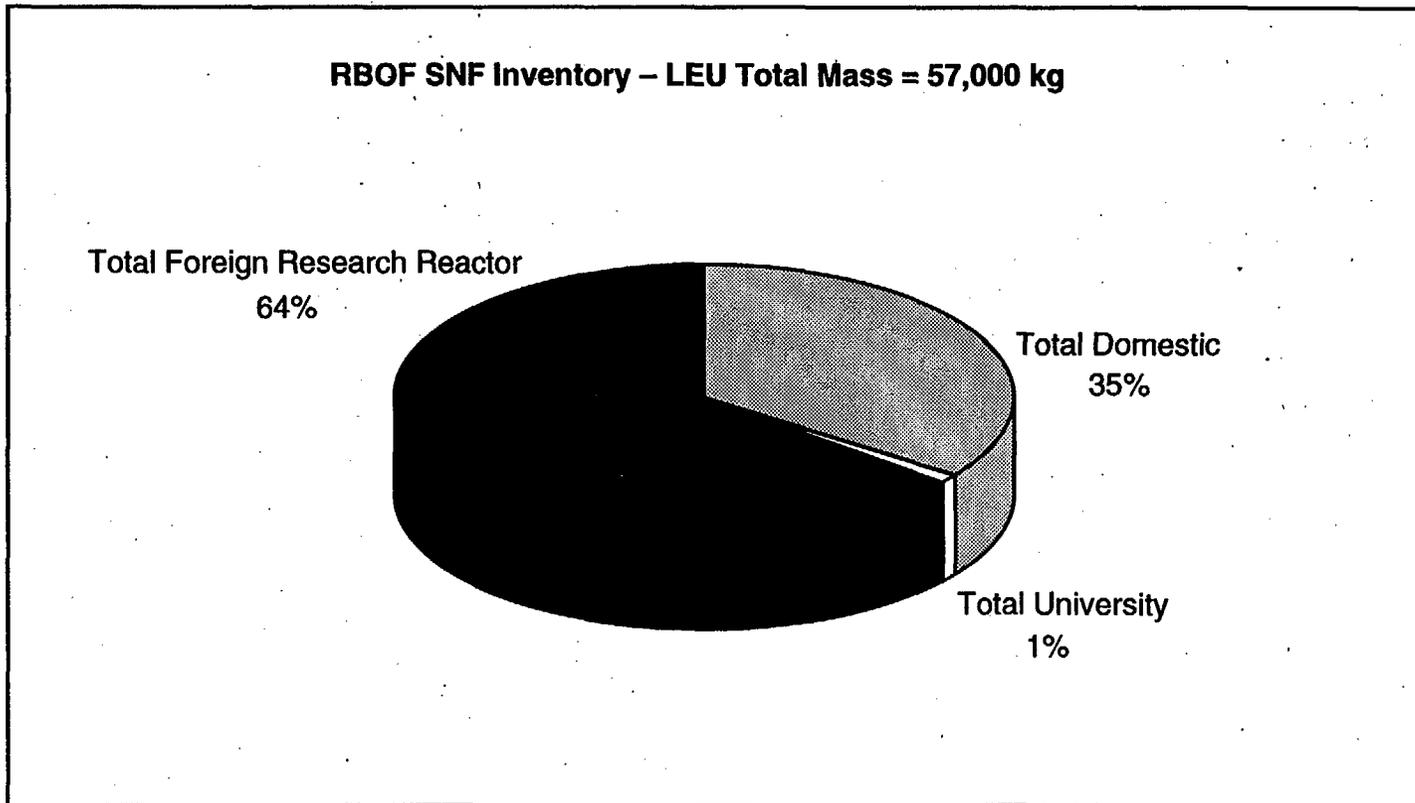


Figure A-7

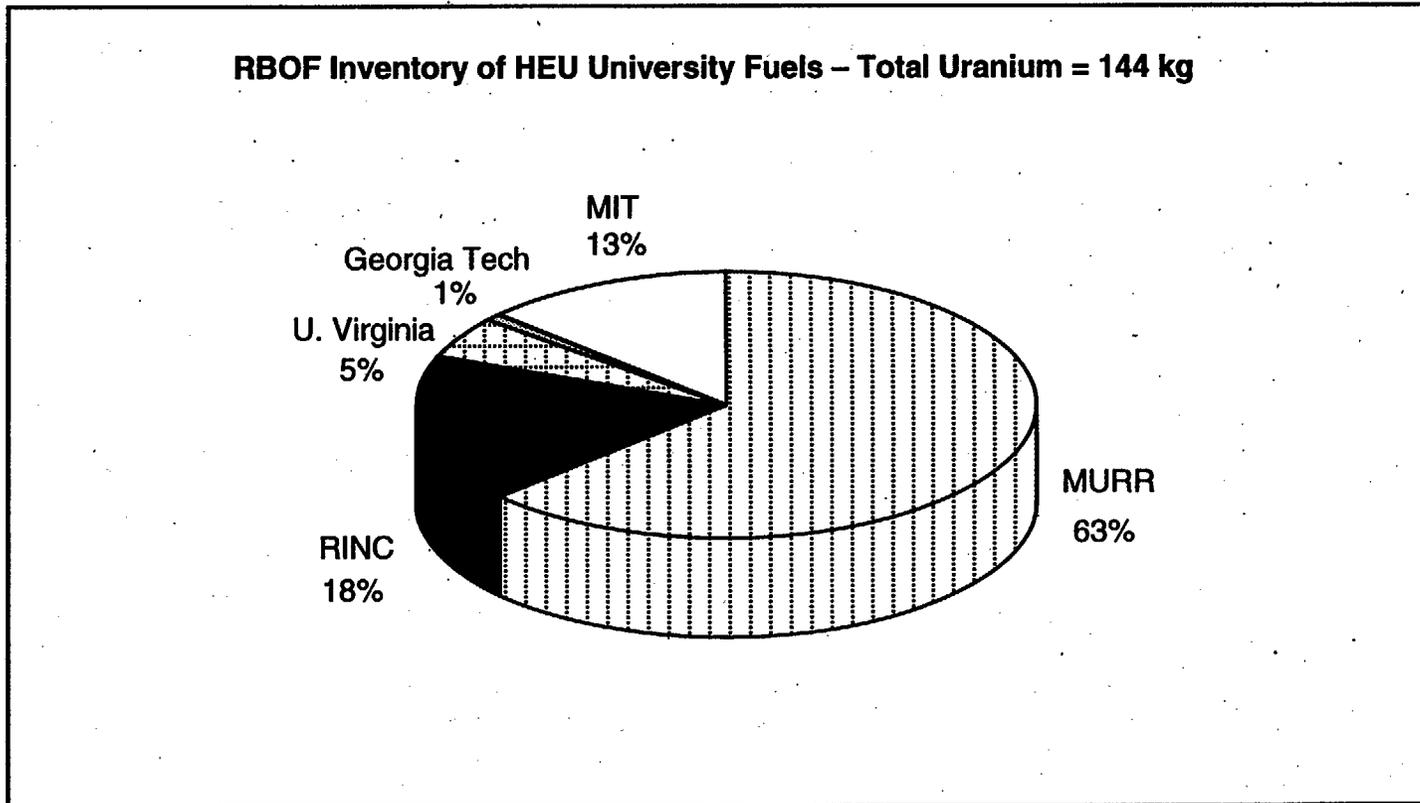


Figure A-8

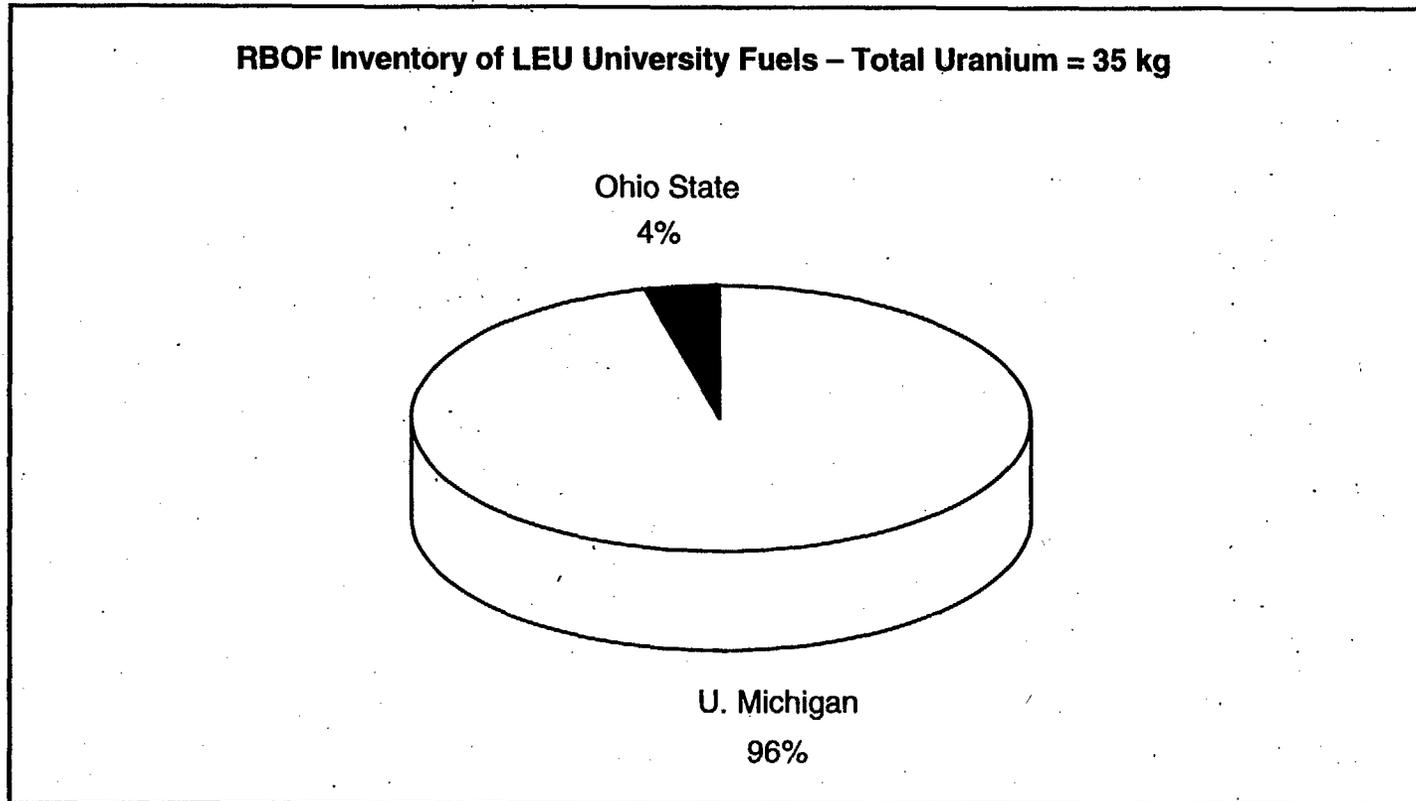
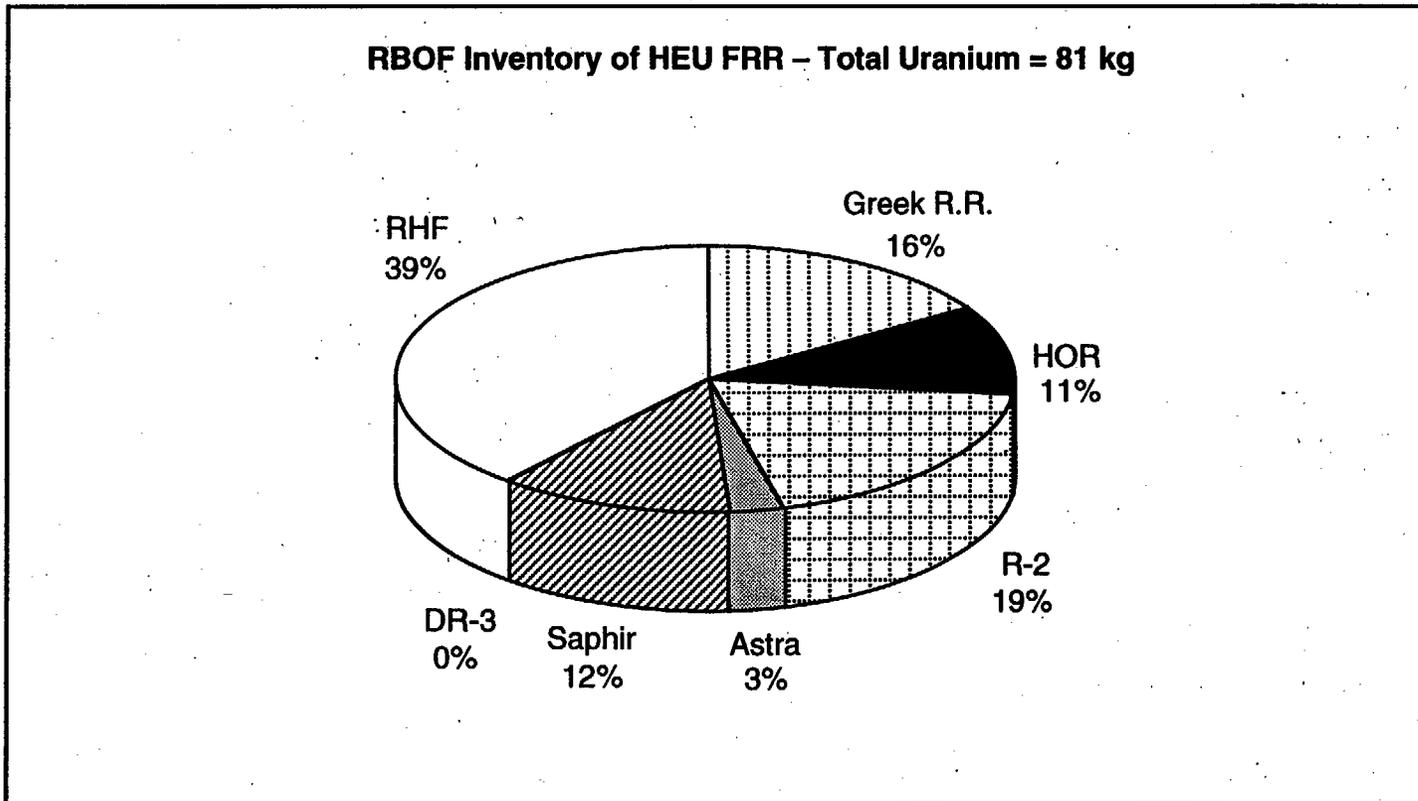
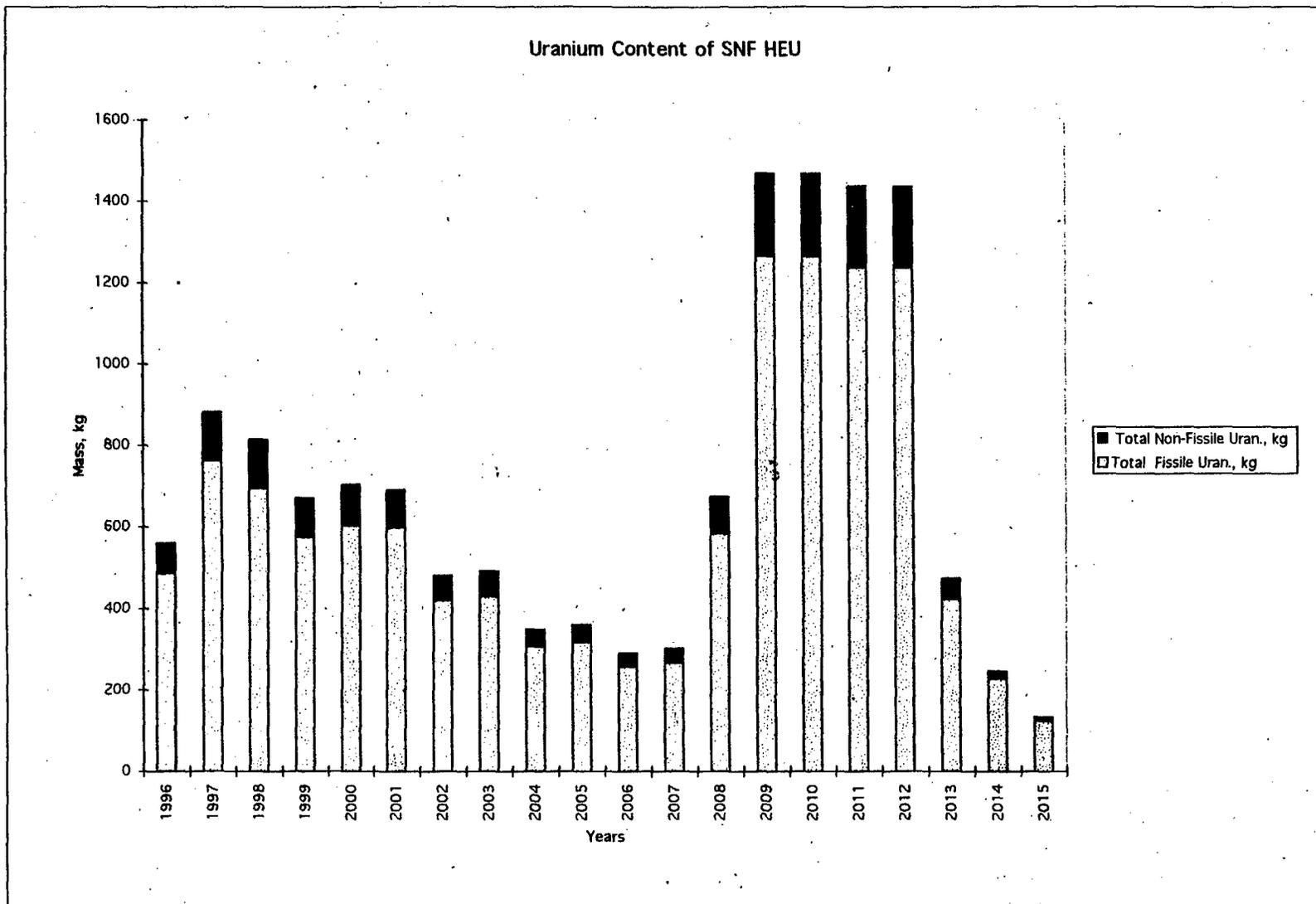
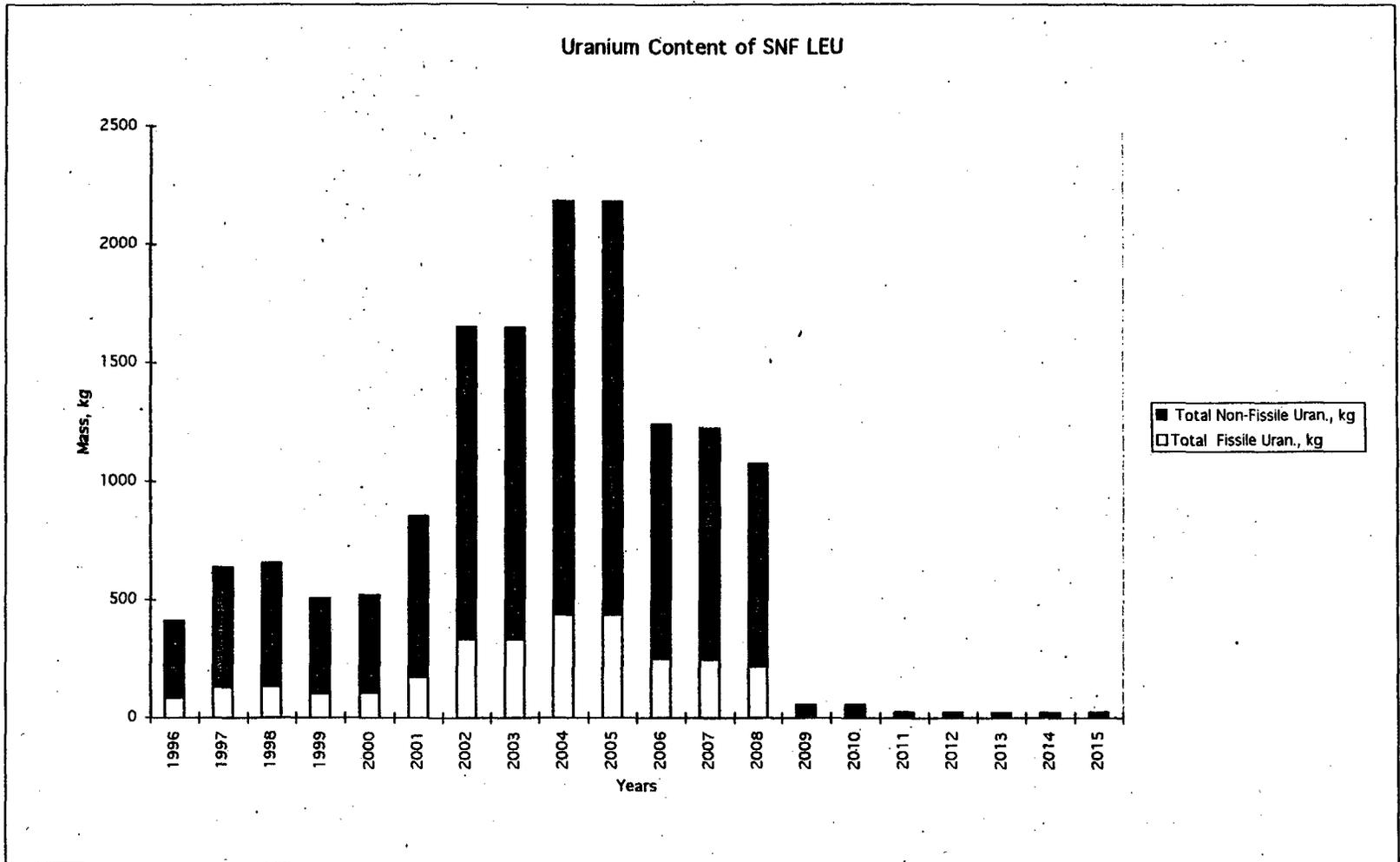


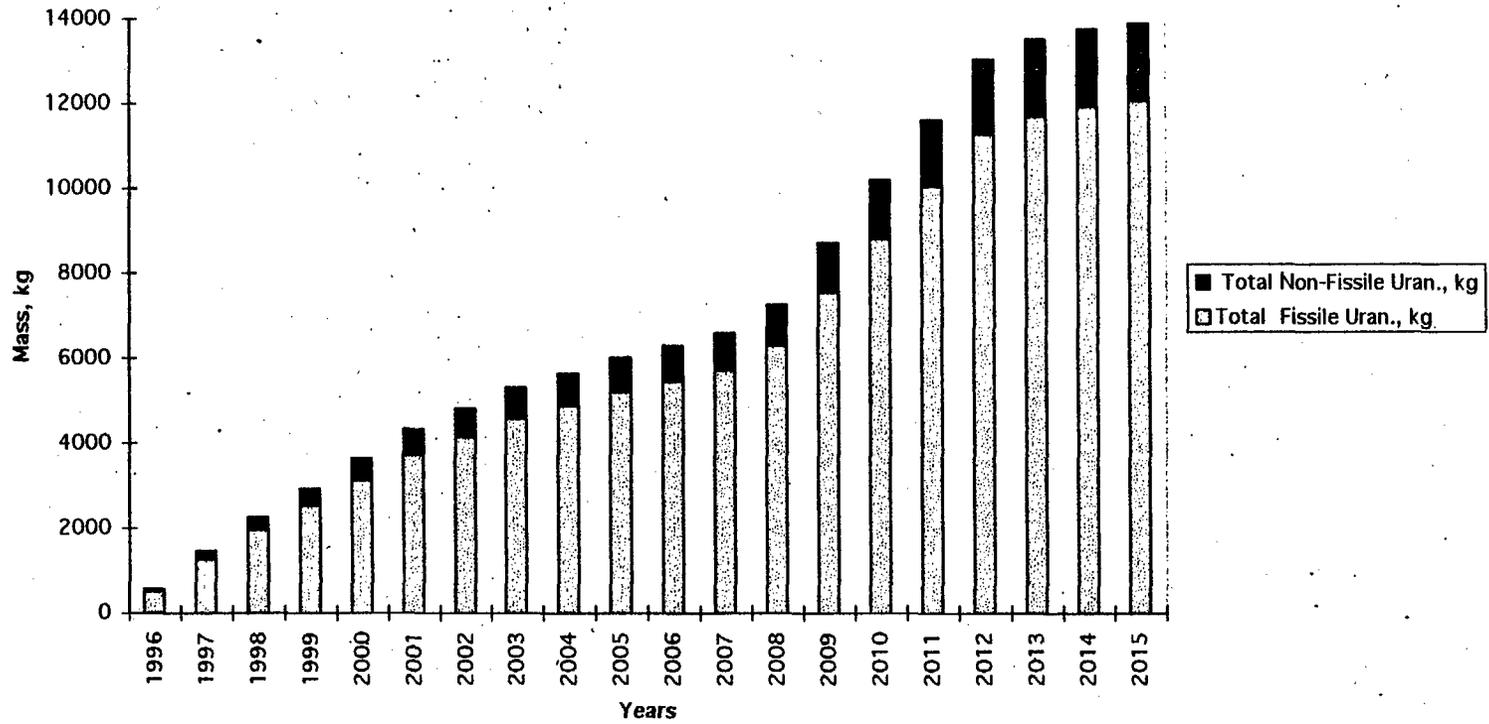
Figure A-9



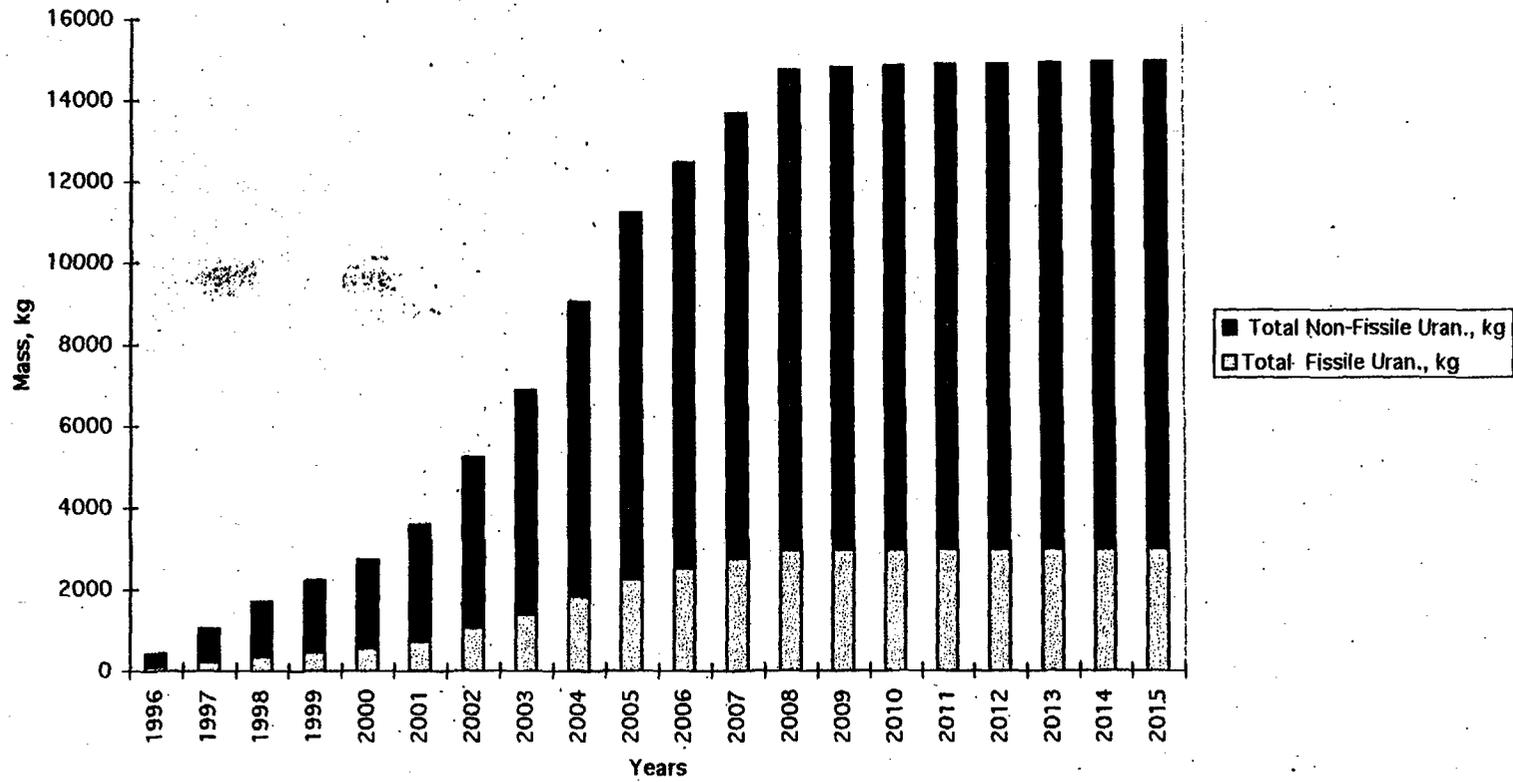


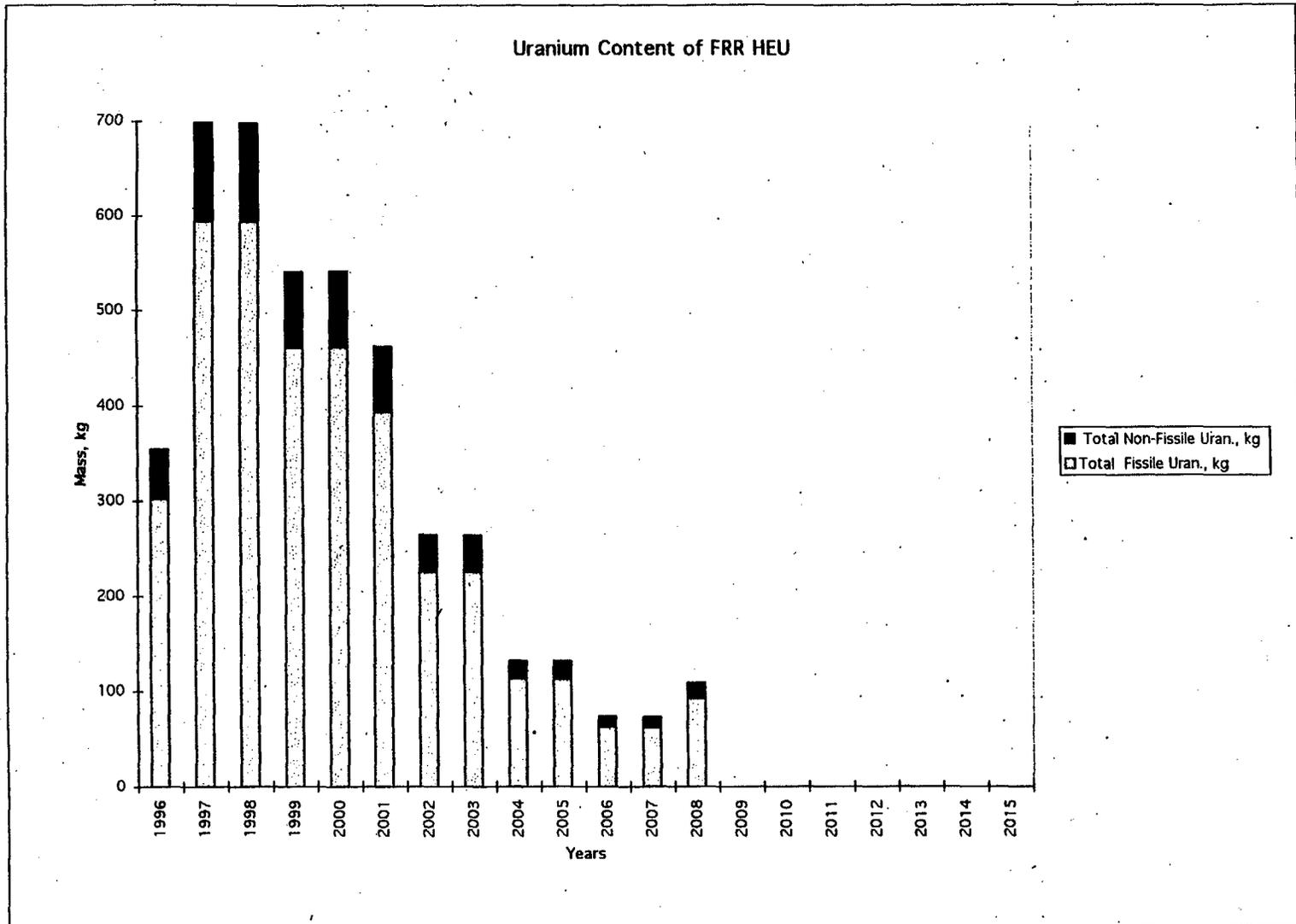


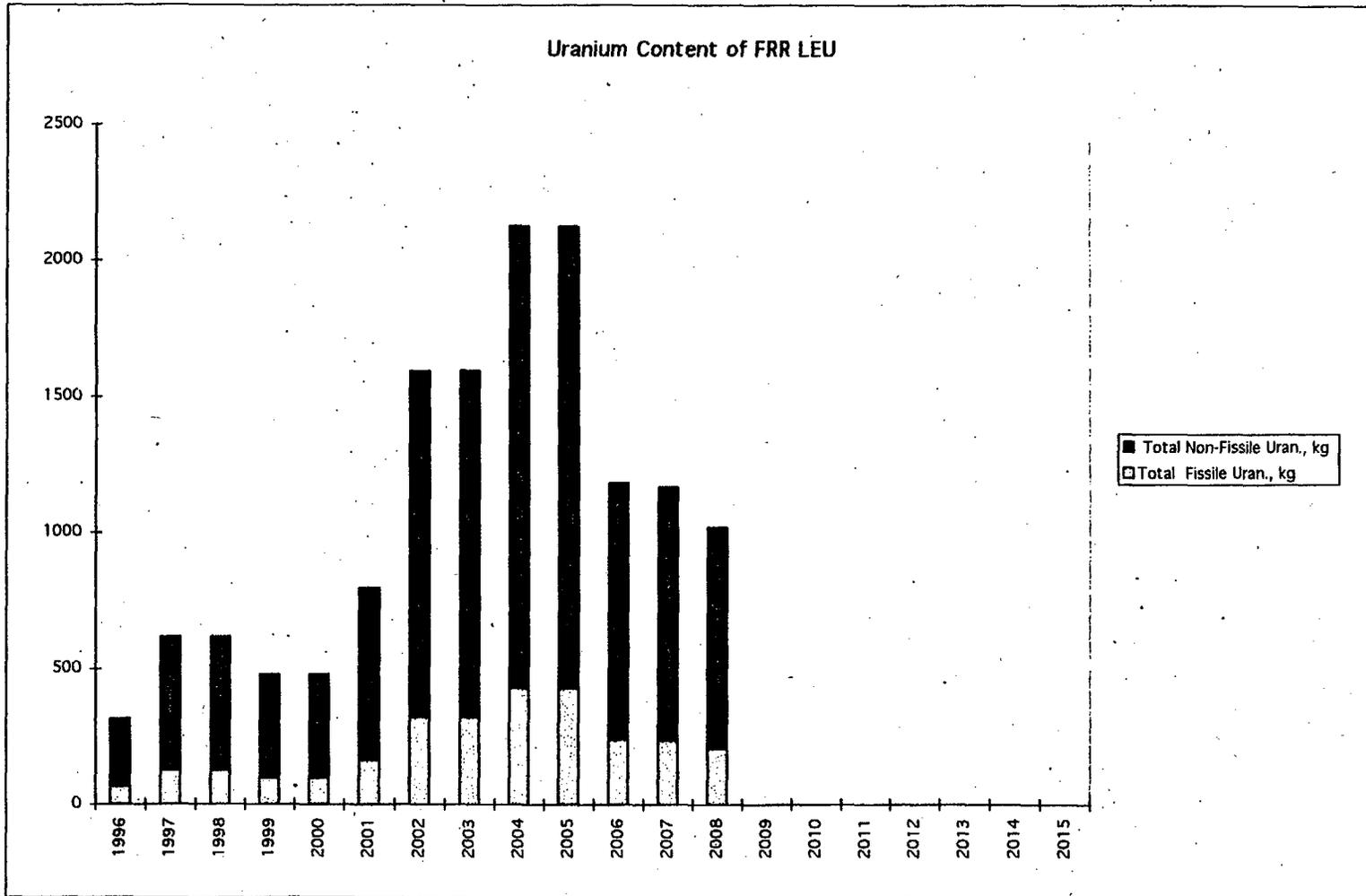
Cumulative Uranium Content of HEU SNF



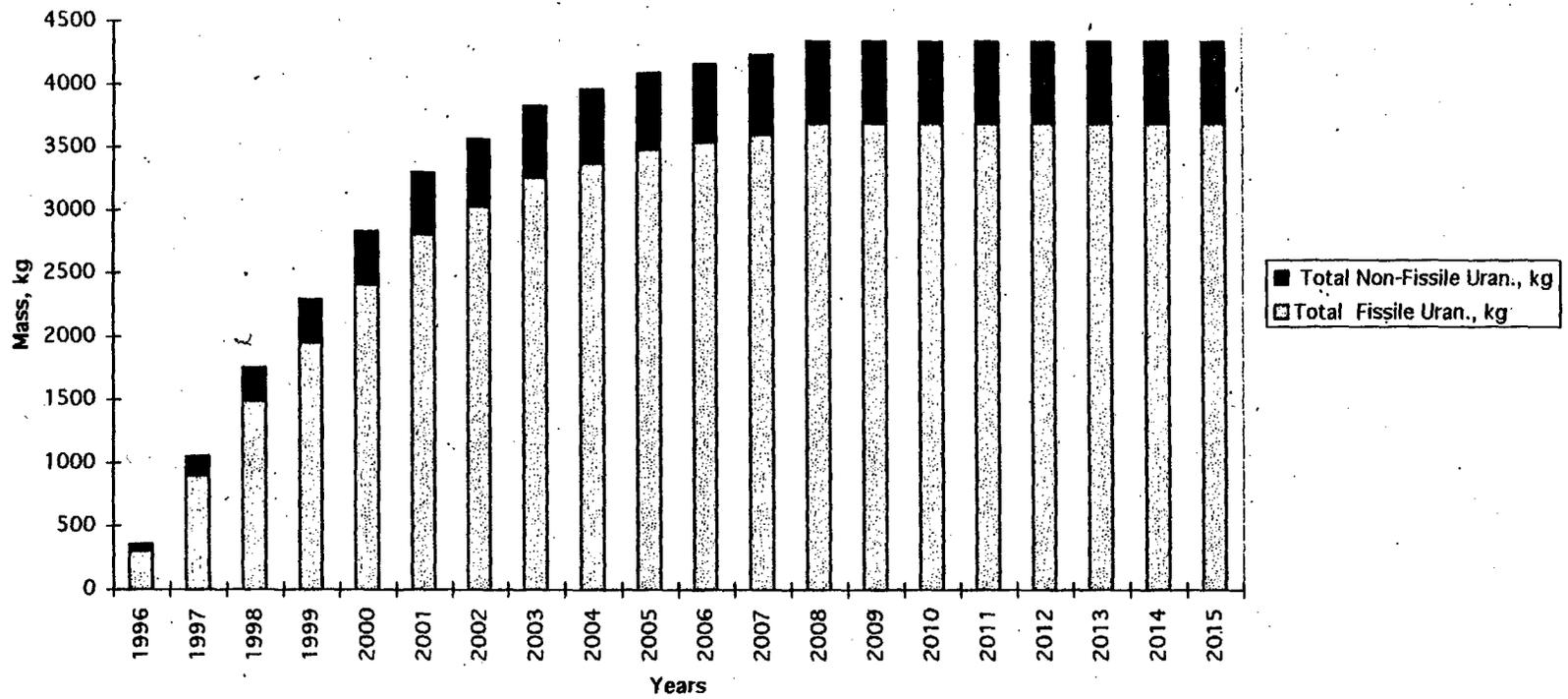
Cumulative Uranium Content of LEU SNF



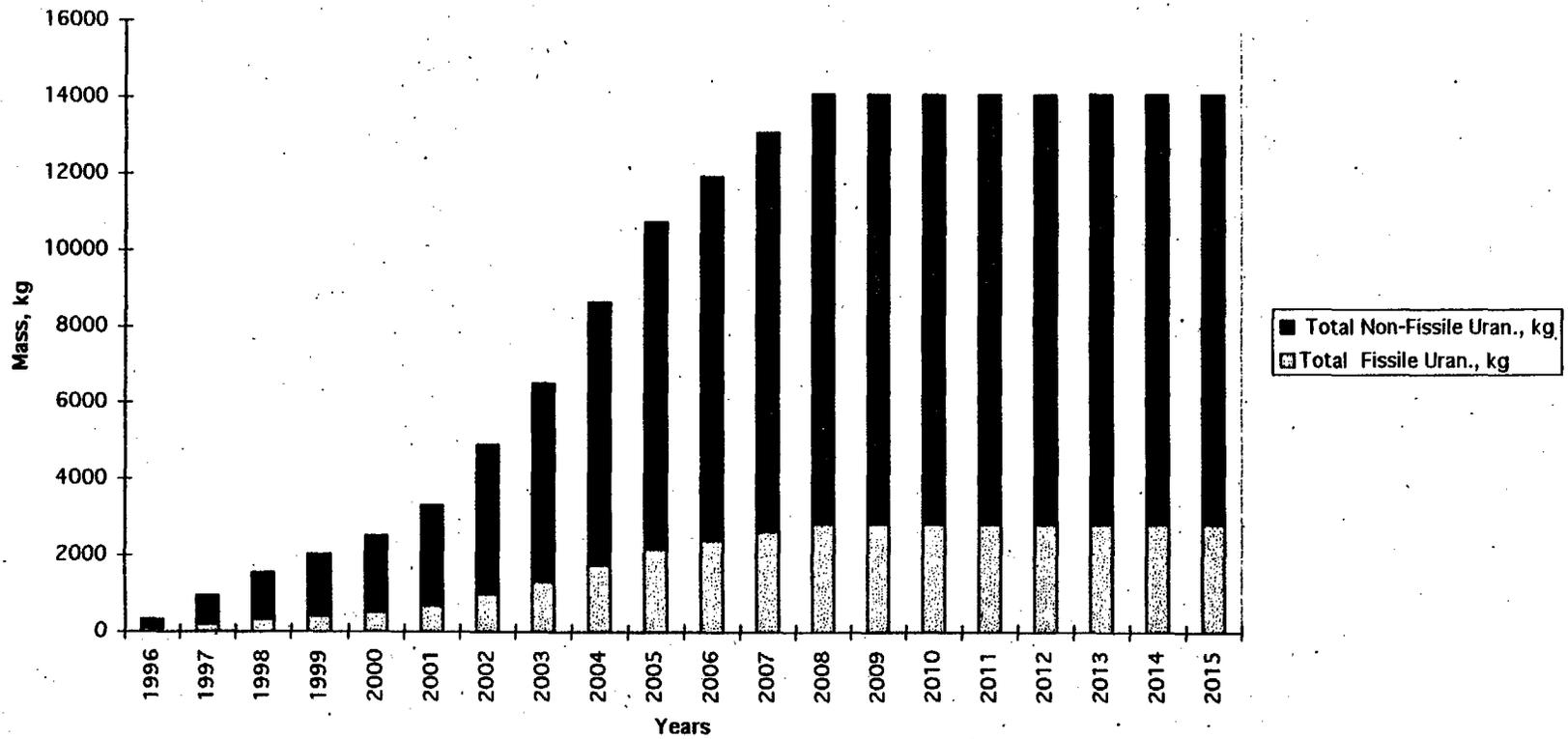




Cumulative Uranium Content of FRR HEU



Cumulative Uranium Content of FRR LEU



Appendix B
Criticality Control Bases for a Repository Licensed Under 10 CFR Part 60

Repository Evaluation Team:

Phyllis M. Lovett, TRW Environmental Safety Systems

Hugh A. Benton, Framatome Cogema Fuels

Thomas P. McLaughlin, Los Alamos National Laboratory

D. Kent Parsons, Los Alamos National Laboratory

Ronald G. Ballinger, Massachusetts Institute of Technology

Appendix B
Criticality Control Bases for a Repository Licensed Under 10 CFR Part 60

B-1. Overview

The purpose of the criticality analyses described herein is to provide a basis for developing comparative cost estimates for various treatment technologies being considered for the disposal of aluminum-based research reactor spent nuclear fuel (SNF). It is not intended that this analysis be used for any other purpose.

Aluminum-based research reactor SNF has three major differences in comparison with commercial SNF that may create special long-term criticality control concerns:

- (1) a higher initial enrichment,
- (2) a lower burnup, and
- (3) an aluminum alloy fuel matrix that is susceptible to rapid corrosion.

During storage, transportation, and repository pre-closure operations and during the repository post-closure period prior to a breach of the waste package, the SNF is in sealed canisters, casks, or waste packages that ensure a dry atmosphere, maintain a fixed geometry, and ensure that any neutron absorber materials remain in the proper locations. Under these conditions, the criticality control strategy for aluminum-based research reactor SNF can easily and economically provide large safety margins. However, after the waste package is breached during the repository post-closure period, the waste package and SNF degrade, geometry control is lost and the fissile nuclides may become separated from the neutron absorber nuclides. Under these post-closure conditions, criticality control for aluminum-based research reactor SNF would depend on using either or both of the following strategies:

- (1) limiting the mass of fissile material within each waste package in order to disperse the highly enriched uranium (HEU) and reduce the probability of assembling a critical mass; and/or
- (2) reducing the reactivity of the fissile material by isotopically diluting the HEU with natural or depleted uranium. Isotopic dilution lowers the average enrichment of the uranium in the waste package and permits loading a larger fissile mass in each waste package, thus it reduces the number of waste packages required to dispose of the research reactor SNF.

The degree to which these measures will be needed will primarily depend on the regulation for criticality control in the repository. The current regulation for criticality control, 10 CFR Part 60.131(b)(7), states: "All systems for processing, transporting, handling, storage, retrieval,

emplacement, and isolation of radioactive waste shall be designed to ensure that a nuclear criticality accident is not possible unless at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety. Each system shall be designed for criticality safety under normal and accident conditions. The calculated effective multiplication factor (k_{eff}) must be sufficiently below unity to show at least a 5% margin, after allowance for the bias in the method of calculation and the uncertainty in the experiments used to validate the method of calculation." A conservative interpretation of this deterministic regulation, seems to require a large criticality safety margin at all times, including during the post-closure waste isolation period, and also prohibits criticality in the repository absent two unlikely, independent events. Given the extremely long time periods of regulatory concern for a geologic repository, the DOE recommended to the Nuclear Regulatory Commission (NRC) that a risk-based demonstration of compliance with criticality control requirements is the only appropriate approach for the post-closure period. However, there is a significant risk that such an approach will not be considered in a licensing proceeding to comply with the current regulation. The DOE has therefore begun a dialog with the NRC that seeks to revise the disposal criticality regulations to permit the use of risk-based analysis.^{1,2}

If the recommended changes are adopted, a risk-based evaluation of criticality and its consequences would be acceptable to the NRC for evaluating the performance of the repository, post-closure. The implementation of a risk-based methodology could alter the current criticality control strategy and could result in cost savings while still meeting repository performance criteria. Recent studies focusing on the geologic disposal of HEU SNF and HLW indicate that the probability of occurrence of a criticality event within a repository is quite small (2×10^{-3} in the first 10,000 years) and that the dose consequences from such events would be negligible (<1%).^{3,4} This result indicates that a risk-based analysis may provide a basis for raising fissile values or decreasing the need for isotopic dilution which would lower the number of disposal packages thereby lowering overall cost. However, the risk-based disposal analysis methodology was not used in this report.

B-2. Criticality Control Strategies

The criticality control strategies for aluminum-based research reactor SNF for the pre-closure period of the repository, and including storage and transportation periods, generally rely upon one or more of the following methods to achieve the desired safety margin:

- Neutron absorber materials
- Geometry control (assembly separation)
- Lack of moderator present (albeit the analyses assume flooded conditions in calculating the safety margin)

With aluminum-based fuel, the criticality control strategies for the post-closure period of the repository do not rely on any of the above pre-closure criticality control methods and must

utilize one or more of the following methods to achieve the desired safety margin:

- Controlling the maximum mass of fissile material in individual waste packages
- Diluting HEU with natural uranium or depleted uranium (i.e., U-238) or other materials
- Leaching, dissolution and colloid formation rates of fissile material and neutron poison
- Hydrological transport and geochemical precipitation of fissile material and neutron poisons

B-3. Fissile Mass Loading for Individual Waste Packages

B-3.1. Fissile Mass Loading Limits Based on Criticality Inside the Waste Package

B-3.1.1. Bases and General Assumptions

The criticality analyses described herein was conducted to support the development of a basis for comparative cost estimates for various treatment technologies being considered for the disposal of aluminum-based research reactor SNF. The results of the analyses are not considered to be design values. The scope of the analyses was limited to just a few conservative cases, primarily due to schedule and funding limitations. The focus of the analyses was on the repository post-closure period after the waste package has been breached and the SNF has become severely degraded while the waste package structure remains in fair condition.

The full range of enrichments contained in the aluminum-based research reactor SNF was grouped into three groups:

- HEU - enrichments greater than or equal to 20 wt% U-235
- LEU - enrichments between 2 wt% and 20 wt% U-235
- VLEU - enrichments less than or equal to 2 wt% U-235

Criticality analyses were performed only for the HEU group and the LEU group; the VLEU group was not analyzed since it does not represent a significant criticality concern and since an estimate of the acceptable mass loading could be inferred from the loading of commercial SNF waste packages.

The calculations assumed that all fuel in the HEU group would be treated as if it contained 93 wt% U-235 and that all fuel in the LEU group would be treated as if it contained 20 wt%

U-235. This approach is conservative but suitable for planning purposes; however, considerable optimization would be needed before actual waste package (or canister) loading begins.

The waste package used in this analysis for the disposal of research reactor SNF is similar to the one planned for commercial SNF, which consists of a 2 cm thick inner barrier of Alloy 825 (high nickel alloy) and a 10 cm thick outer barrier of carbon steel. This approach assures that the waste package for the research reactor SNF will have approximately the same performance life-time as the one used for commercial SNF. As with commercial SNF, criticality control for research reactor SNF configurations is provided by the borated-stainless steel basket within the intact waste package. The neutron absorber material incorporated into the borated-stainless steel basket is designed to ensure an adequate subcritical limit as long as the fuel remains in the intact basket. When the waste package is eventually breached, the intrusion of water is expected to corrode the aluminum matrix of the research reactor SNF long before the stainless-steel basket material. Thus, in order to reach a critical configuration, the fissile material must move out of the basket. Furthermore the boron in the presumably intact basket structure must be removed from the system. The critical mass of such configurations are used herein to define the planning values for fissile content of aluminum-based research reactor SNF in individual waste packages.^a

The fissile mass planning value for any waste package is greatly dependent on the scenario by which (1) the fuel and the waste package degrade, (2) the fissile material is assumed to separate from various neutron absorber materials, and (3) the fissile material accumulates into a low (neutron) leakage configuration that would yield a critical mass.

A U-235 mass planning value for criticality control was developed based on a nominal waste package design that would have a k_{eff} sufficiently below unity to achieve the required safety margin. The NRC regulation for criticality control in the repository (10 CFR 60.131(b)(7)) requires a 5% margin of safety ($k_{\text{eff}} \leq 0.95$) plus additional margin for bias and uncertainty in the criticality calculations. For commercial SNF, without burnup credit, the present best estimate for the bias and uncertainty margin is 2%. Since burnup credit will not be claimed for research reactor SNF,^b a $k_{\text{eff}} \leq 0.93$ was used as the subcritical limit to determine the maximum fissile content of waste packages containing aluminum-based research reactor SNF.

^a The maximum fissile content analysis takes into account the anticipated degradation of aluminum. The calculated per-package fissile loadings identified in this study are for aluminum-based SNF; therefore, they do not apply to more robust waste forms such as commercial SNF, naval reactor SNF, or other nonaluminum-based SNF.

^b Burnup credit for research reactor SNF does not yield as great a benefit in criticality analyses, as it does for commercial SNF, because of its high initial enrichment and its relatively low atom percent burnup (typically less than 20 wt% of the initial U-235 atoms are fissioned, yielding effective final enrichments in the 60-70% range). Furthermore, because of the variety of research reactor SNF types and the variation in their operating histories, obtaining the records and isotopic assays needed to justify burnup credit for research reactor SNF may be impractical. The criticality analyses used to determine the fissile mass loading for waste packages containing aluminum-based research reactor SNF neglected burnup credit and utilized the fissile content of fresh fuel.

Other general assumptions used throughout these criticality analyses are:

- All aluminum-based research reactor SNF assemblies can reasonably be represented in criticality analyses by a typical MTR fuel assembly containing 300 grams of U-235 per assembly.
- The source region was always modeled in the MCNP {Monte Carlo N Particle Transport Code} as a homogeneous mixture containing small particles of UO_2 dispersed and uniformly suspended in water.
- The neutron absorber materials in the waste package basket were ignored as were the corrosion products of the aluminum-based fuel elements and the waste package materials.
- Water reflection at all boundaries.
- Beginning of life fuel loadings were used and fission product poisons were not included.

These assumptions and the following analyses are consistent with those used in the report on the direct disposal of foreign research reactor SNF.⁵

B-3.1.2. Research Reactor SNF Containing High Enriched Uranium (HEU)

The assumptions used to estimate the planning value for the fissile mass of U-235 in HEU that achieves the required subcritical limit with degraded fuel configurations within the waste package are as follows:

- All of the uranium in the fuel was transported to the bottom of the waste package as particles of UO_2 (assuming an enrichment of 93 wt% U-235) and homogeneously suspended in a water slurry. (The homogeneous slurry assumption significantly reduces the acceptable amount of U-235.)
- None of the aluminum corrosion products from the fuel elements were transported to the bottom of the waste package. (The corrosion products would displace water and significantly increase the acceptable amount of U-235.)
- The slurry is conservatively assumed to fill the gap between the cylindrical wall of the horizontal waste package (84 cm I.D.) and the bottom plate of the basket, which results in a maximum slurry depth of 6 cm.

- The top surface of fissile mass is assumed to have water as a neutron reflector and the neutron absorber materials in the bottom plate of the basket was neglected. (Including the bottom plate of the basket would significantly increase the planning value for fissile material.)
- The bottom and sides of the fissile mass are contained by the cylindrical wall of the waste package, i.e., 2 cm of Alloy 825 backed with 10 cm of carbon steel.
- The fuel-water slurry is confined to a length of no less than about one meter, i.e., approximately the length of one fuel assembly, although various lengths were analyzed to approximately optimize the amount of moderation.

The result of the MCNP calculation showed that 14.4 kg U-235 as HEU with 93 wt% enrichment in this configuration would achieve the required subcritical limit, i.e., $k_{\text{eff}} \leq 0.93$. This mass of U-235 represents about 48 MTR type fuel assemblies in a new waste package specifically designed for the HEU fuel.

B-3.1.3. Research Reactor SNF Containing Low Enriched Uranium (LEU)

The planning value for the critical mass of U-235 increases as the uranium enrichment decreases due to the increased neutron absorption in the U-238 and dilution of U-235 by U-238. For low enriched uranium (LEU), i.e., uranium with a U-235 enrichment of 2 wt% to 20 wt%, the critical mass is significantly greater than that for 93 wt% enriched uranium.

The assumptions used to estimate this planning value for the fissile mass of U-235 in LEU that achieves the required subcritical limit with degraded fuel configurations within the waste package are as follows:

- All of the uranium in the fuel is homogeneously suspended at the bottom of the waste package as particles of UO_2 (assuming an enrichment of 20 wt% U-235) homogeneously suspended in a water slurry.
- None of the aluminum corrosion products from the fuel elements were transported to the bottom of the waste package. (The corrosion products would displace water and significantly increase the acceptable amount of U-235.)
- The slurry is conservatively assumed to fill the gap between the cylindrical wall of the horizontal waste package (105.8 cm I.D.) and the bottom plate of the basket, which results in a the maximum slurry depth of 6 cm.
- The top surface of fissile mass is assumed to have water as a neutron reflector (the reduction in k_{eff} due to the neutron absorber materials in the plate of the basket was neglected).

- The bottom and sides of the fissile mass are contained by the cylindrical wall of the waste package, i.e., 2 cm of Alloy 825 backed with 10 cm of carbon steel.
- The fuel-water slurry is confined to a length of no less than about one meter, i.e., approximately the length of one fuel assembly, although various lengths were analyzed to approximately optimize the amount of moderation.

The MCNP calculations showed that a $k_{\text{eff}} \leq 0.93$ was achieved for LEU with 43 kg U-235 in a standard waste package designed for 12 commercial Pressurized Water Reactor (PWR) fuel assemblies. The actual k_{eff} was only 0.79 but the waste package was full with 144 MTR type fuel assemblies. The next larger standard size waste package (designed for 21 commercial PWR fuel assemblies) was too large to meet the $k_{\text{eff}} \leq 0.93$ criteria. Clearly, an optimized waste package design could accommodate a somewhat larger fissile loading.

If depleted uranium in the same or a similar chemical form as the U-235 in the research reactor SNF is used to isotopically dilute the HEU in individual waste packages to an average uranium enrichment of less than 20 wt% U-235, the LEU mass planning values for the U-235 content per waste package can be used since the U-235 in the HEU and the U-238 in the depleted uranium will be dissolved and transported at the same rates. If the chemical forms are significantly different, the full benefit of the isotopic dilution may not be achieved and a more detailed analysis of the dissolution scenario will be needed to assess the impacts on the critical mass limits for that waste form.

B-3.1.4. Research Reactor SNF Containing Very Low Enriched Uranium (VLEU)

For very low enriched uranium (VLEU), i.e., uranium with a U-235 enrichment of less than or equal to 2 wt%, the critical mass is significantly greater than for LEU. The critical mass within the waste package was not calculated specifically for VLEU, it was inferred from the criticality analyses performed for commercial SNF which show that large waste packages of commercial SNF with an effective final enrichment of about 2 wt% U-235 can safely contain 200 kg of fissile material.

B-3.2. Fissile Mass Loading Limits Based on Criticality Outside the Waste Package

The assumptions used to estimate the planning value for the fissile mass of U-235 in HEU that achieves the required subcritical limit with degraded fuel configurations outside the waste package are as follows:

- All of the uranium in the fuel collect in a cylindrical configuration in the fine gravel beneath the waste package as a homogeneous dispersion of fine particles of UO_2 (with an enrichment of 93 wt% U-235). The diameter and length of the cylinder were varied within the 60 cm depth of the drift invert to find an approximate optimum configuration. This geometry assumes that the fissile material was homogeneously concentrated into a smaller volume than existed in

the original waste package. This is very conservative since the fissile material will naturally spread and distribute heterogeneously as it moves away from its original location.

- The natural elemental composition of the significant neutron absorbers in the tuff gravel is: Na (2.65 wt%), K (3.91 wt%), Ca (0.4 wt%), Fe (0.45 wt%), and Mg (0.28 wt%) but did not include the trace amounts of rare earth elements that can significantly reduce reactivity.
- None of the neutron poison from the waste package were assumed to have migrated into the tuff.
- The tuff gravel is assumed to contain 14% water (by volume) distributed uniformly throughout the drift invert (floor). Forty-three percent of this water is in the pores of the tuff and the remaining 57% is on surfaces or in the void spaces between particles of tuff. The amount of water on surfaces or in the void spaces is consistent with the maximum value used in the *Draft Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel*.⁶

The result of the MCNP calculation showed that 7.8 kg U-235 as HEU with 93 wt% enrichment would achieve the required subcritical limit, i.e., $k_{\text{eff}} \leq 0.93$. The accumulation of fissile material in a near optimal configuration within the invert was considered too conservative for the purposes of the planned cost comparisons since the engineered barrier system could be designed to prevent accumulation in optimal configurations. Therefore, the 7.8 kg value was not used for the cost evaluation of the alternative treatment technologies for the disposal of aluminum-based research reactor SNF.

B-4. Summary of Maximum Planning Value for Fissile Mass in One Waste Package^c

The criticality analyses that support the planning values for the fissile mass loading, as presented in the preceding sections, contain many conservative assumptions; therefore, they are recommended only for use in planning and costing studies, and are not intended for use as design values. Some of the assumptions that provide this conservatism are summarized below. The detailed assumptions are provided in Section B.3.

- Only the maximum enrichment of each group (i.e., 93 wt%, 20 wt%, and 2 wt%) was used to calculate the fissile mass required to produce a $k_{\text{eff}} \leq 0.93$.

^c The analysis takes into account the anticipated degradation of aluminum. The calculated per-package fissile loading identified in this study for aluminum-based SNF, therefore, do not apply to more robust waste forms such as commercial SNF, naval reactor SNF, or other nonaluminum-based SNF.

- All of the fissile material in the waste package was assumed to separate completely from the neutron absorber materials and corrosion products.
- The assumed geometry outside the waste package requires the fissile material to concentrate into a small homogeneous volume. It is more likely that the fissile material will naturally spread and disperse heterogeneously as it moves away from its original location.
- Depletion of U-235 and buildup of fission product poisons due to burnup was neglected.

Considering the magnitude of conservatism imbedded in these assumptions for criticality, the planning values for fissile mass loading based on criticality within the waste package were judged to provide a reasonable basis for developing comparative cost estimates for various treatment technologies being considered for the disposal of aluminum-based research reactor SNF. These planning values for fissile mass loadings, which are summarized in Table B-1, were selected as the basis for evaluating the relative merits of the nine treatment technologies for aluminum-based research reactor SNF.

Table B-1. Summary Planning Values for the Fissile Mass Loading per Waste Package^d

Research Reactor SNF Enrichment	Selected Value for Evaluation of Technologies (kg U-235)
HEU (≥ 20 wt% to 93 wt% U-235)	14.4
LEU (2 wt% to 20 wt% U-235)	43
VLEU (≤ 2 wt% U-235)	200

^d For use in evaluating the alternative treatment technologies for research reactor SNF in a repository licensed under 10 CFR Part 60.

B-5. Future Optimization of the Waste Package Fissile Mass Loading

The planning values for the fissile mass loadings shown in Table B-1 can achieve the subcritical limit required by the current NRC regulations for disposal of high-level radioactive wastes in geologic repositories (10 CFR Part 60.131(b)(7)). The DOE has provided comments to the NRC recommending that the new regulations focus on the waste isolation performance of the total repository system (rather than on the performance of various components or subsystems) and that the new regulations also explicitly permit the use of probabilistic risk assessment methodologies (as opposed to only deterministic analyses) to demonstrate compliance with the repository performance requirements of the regulations.^{1,2} If the NRC incorporates DOE's recommended changes into the new regulations, more cost effective design solutions will become available. At that time, an evaluation of the merits of isotopic dilution and/or the most cost effective amount of dilution, using either natural or depleted uranium, should be included.

B-6. References

1. *Comments on Proposed Changes to 10 CFR Part 60 Related to Design Basis Events*, Letter from R.A. Milner (RW-30) to U.S. Nuclear Regulator Commission, Secretary, Docketing and Service Branch, June 16, 1995.
2. *U.S. Department of Energy Annotated Outline for Topical Report, "Disposal Criticality Analysis,"* Letter from Stephan J. Brocoum (YMSCO) to Michael J. Bell (U.S. Nuclear Regulatory Commission), April 12, 1996.
3. *Performance Assessment of the Direct Disposal in Unsaturated Tuff of Spent Nuclear Fuel and High-Level Waste Owned by the U.S. Department of Energy*, SAND94-2563, Sandia National Laboratories, March 1995, p. ES-6.
4. *Consideration of Criticality when Directly Disposing of Highly Enriched Spent Nuclear Fuel in Unsaturated Tuff: Bounding Estimates*, SAND96-0866, Sandia National Laboratories, May 1996
5. *Assessment of Technical Issues and Cost for Direct Disposal of Foreign Research Reactor SNF*, A00000000-01717-5705-00010, Revision 0, TRW Environmental Safety Systems, December 1995.
6. *Draft Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel*, DOE/EIS-0218D, March 1995.

Appendix C
Cost Evaluation Model and Analysis

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Appendix C
Cost Evaluation Model and Analysis

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1.0 Objective

The objective of the cost evaluation is to identify major cost differences among the technologies and develop an estimate of life cycle cost for each technology.

2.0 Approach

The overall approach to cost evaluation is to develop high level comparative cost estimates, using pre-conceptual design information, and based upon estimates and historical values from comparable facilities to the maximum extent possible. Bottom up estimates are not essential for this comparative cost evaluation, though they may be performed for more detailed study of options selected for further study. Though the precision of any given cost may be rough, because of consistent treatment between technologies, the relative ranking of the options is reasonable. The focus in this comparative evaluation is on consistency and identification of major cost drivers.

The cost models described in this Appendix are presented in Section 4.2 of the main report. Some material is repeated in the interest of continuity and minimizing the need for cross-referencing. Two different types of costs are presented and discussed. Conceptual costs, which do not include allowances for uncertainties, are presented in Section 4.2.2 of the main report and are the subject of Sections 7.0, 8.0 and 10.0 of this Appendix C. Uncertainties associated with conceptual costs are treated in Section 4.2.3 of the main report and Section 9.0 of this Appendix. Finally, Section 4.2.4 of the main report, Section 5.0 and Section 11.0 of this Appendix address cost comparison points which include appropriate adjustment for uncertainty.

Total life cycle costs are included through 2035, with an additional allowance for decontamination and decommissioning of facilities. All costs are presented in 1996 dollars, and are shown in the year of actual expenditure.

3.0 Assumptions

- Table C3.0-1 shows the assumed deliveries of SNF shipping casks to SRS on an annual basis. Table C3.0-2 shows the assumed deliveries of SNF assemblies. Figures C3.0-1 through C3.0-3 show the same information in graphical format. (S.W. O'Rear, WSRC)
- Table C3.0-3 presents assumptions used in the analysis of material handling functions for all treatment technologies.
- Table C3.0-4 shows the assumed number of packages for each treatment technology.
- It is assumed that the repository will begin operations in 2015, and that the first shipment of material from research reactor SNF will occur in 2020.

Table C3.0-1 Assumed Aluminum-Based SNF Shipping Cask Deliveries to SRS

YEAR	CASK DELIVERIES							
	FRR LEU CASKS	FRR HEU CASKS	TOT FRR CASKS	DRR LEU CASKS	DRR HEU CASKS	TOT DRR CASKS	INEL HEU CASKS	HFIR HEU CASKS
1995	1	3	4	1	6	7		
1996	5	23	28	2	17	19		12
1997	11	49	60	2	14	16		14
1998	11	49	60	2	19	21		
1999	11	49	60	1	12	13		5
2000	11	49	60	3	23	25		4
2001	22	38	60	3	23	25		12
2002	37	23	60	3	23	25		12
2003	37	23	60	3	23	25		12
2004	50	10	60	3	23	25		12
2005	50	10	60	3	23	25		12
2006	29	6	35	3	23	25		12
2007	29	5	34	3	23	25		12
2008	24	10	34	2	22	24	11	12
2009				2	22	24	34	24
2010				2	22	24	34	24
2011				1	9	10	34	24
2012				1	9	10	34	24
2013				1	9	10	6	24
2014				1	9	10		22
2015				1	9	10		11
2016				1	9	10		6
2017				1	9	10		
2018				1	9	10		
2019				1	9	10		
2020				1	9	10		
2021				1	9	10		
2022				1	9	10		
2023				1	9	10		
2024				1	9	10		
2025				1	9	10		
2026				1	9	10		
2027				1	9	10		
2028				1	9	10		
2029				1	9	10		
2030				1	9	10		
2031				1	9	10		
2032				1	9	10		
2033				1	9	10		
2034				1	9	10		
2035				1	9	10		

Table C3.0-2 Assumed Deliveries of Aluminum-Based SNF Assemblies to SRS

YEAR	ASSEMBLIES								
	FRR LEU (MTRE)	FRR HEU (MTRE)	FRR TOT (MTRE)	DRR LEU (MTRE)	DRR HEU (MTRE)	DRR TOT (MTRE)	INEL HEU (MTRE)	HFIR ASSY (not MTRE)	RHF ASSY (not MTRE)
1995	28	125	153	10	89	99			4
1996	196	891	1087	34	310	344		12	6
1997	386	1757	2143	16	140	156		14	6
1998	385	1756	2141	35	311	345			8
1999	299	1362	1661	25	223	248		5	8
2000	299	1364	1663	40	360	400		4	6
2001	615	1046	1661	40	360	400		12	8
2002	1014	649	1663	40	360	400		12	6
2003	1013	648	1661	40	360	400		12	8
2004	1397	266	1663	40	360	400		12	6
2005	1395	266	1661	40	360	400		12	8
2006	775	148	923	40	360	400		12	6
2007	765	146	911	40	360	400		12	8
2008	645	264	909	38	346	384	502	12	2
2009				38	346	384	1552	24	
2010				38	345	383	1552	24	
2011				15	135	150	1552	24	
2012				15	135	150	1552	24	
2013				15	135	150	276	24	
2014				15	135	150		22	
2015				15	135	150		11	
2016				15	135	150		6	
2017				15	135	150			
2018				15	135	150			
2019				15	135	150			
2020				15	135	150			
2021				15	135	150			
2022				15	135	150			
2023				15	135	150			
2024				15	135	150			
2025				15	135	150			
2026				15	135	150			
2027				15	135	150			
2028				15	135	150			
2029				15	135	150			
2030				15	135	150			
2031				15	135	150			
2032				15	135	150			
2033				15	135	150			
2034				15	135	150			
2035				15	135	150			

ASSUMPTIONS AND NOTES:

1. Total actual FRR and RHF assemblies are 17,781 as shown in the FRR SNF EIS (w/o MTRE adjustment)
2. Total DRR/DOE, INEL, and HFIR assemblies are the 8,006 option 4a items listed in the SNF/INEL PEIS through 2005 plus 150/yr. more assemblies for 2006-2035.
3. MTRE means 5 assemblies will fit into SRS General Purpose Tube, Square Can, or L-Basin Can
4. FRR assemblies are actual thru '95.96-98 est. @ 33/cask and '99 on est. @ 25/cask (w/o MTRE adjustment)
5. DRR/DOE assembly receipt rate based on W. D. Clark 4/17/95 schedule thru FY'99.FY 2000 on est. @ 16/cask
6. NIST Operates thru 2024 @ 30 ASSY/YR, HFIR operates thru 2015 @ 12/YR (170 cores are shown to be on hand in the PEIS data thru 2005), HRBR operates indef. @ 63-80/yr.
7. FRR LEU/HEU Splits are estimated from 12/94 Matos Tabulation
8. DRR LEU/HEU split estimated at 10% LEU
9. Current RBOF inventory is assumed to be 2752 assemblies.

Figure C3.0-1 Assumed Cask Deliveries to SRS

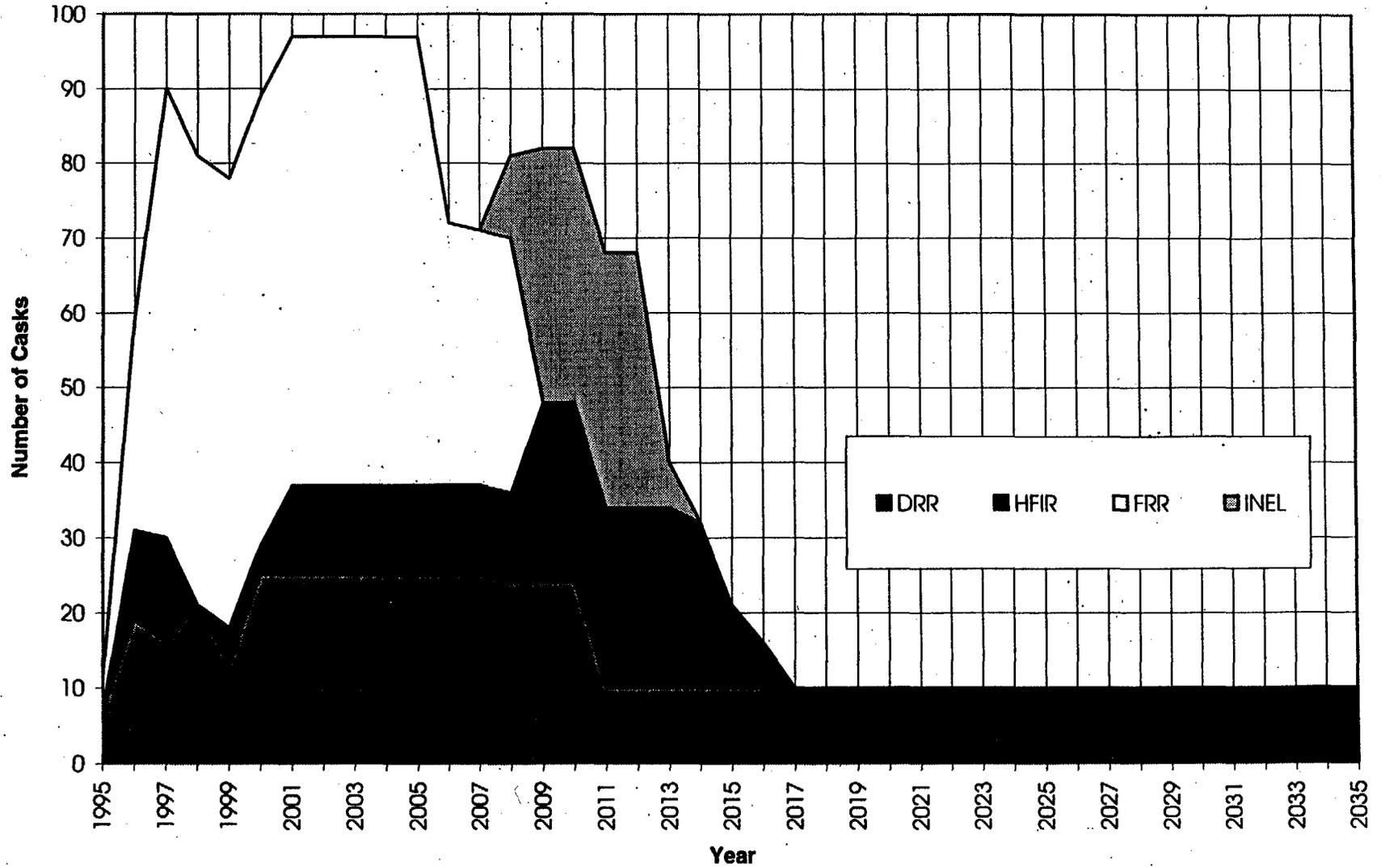


Figure C3.0-2 Assumed MTRE SNF Assembly Deliveries to SRS

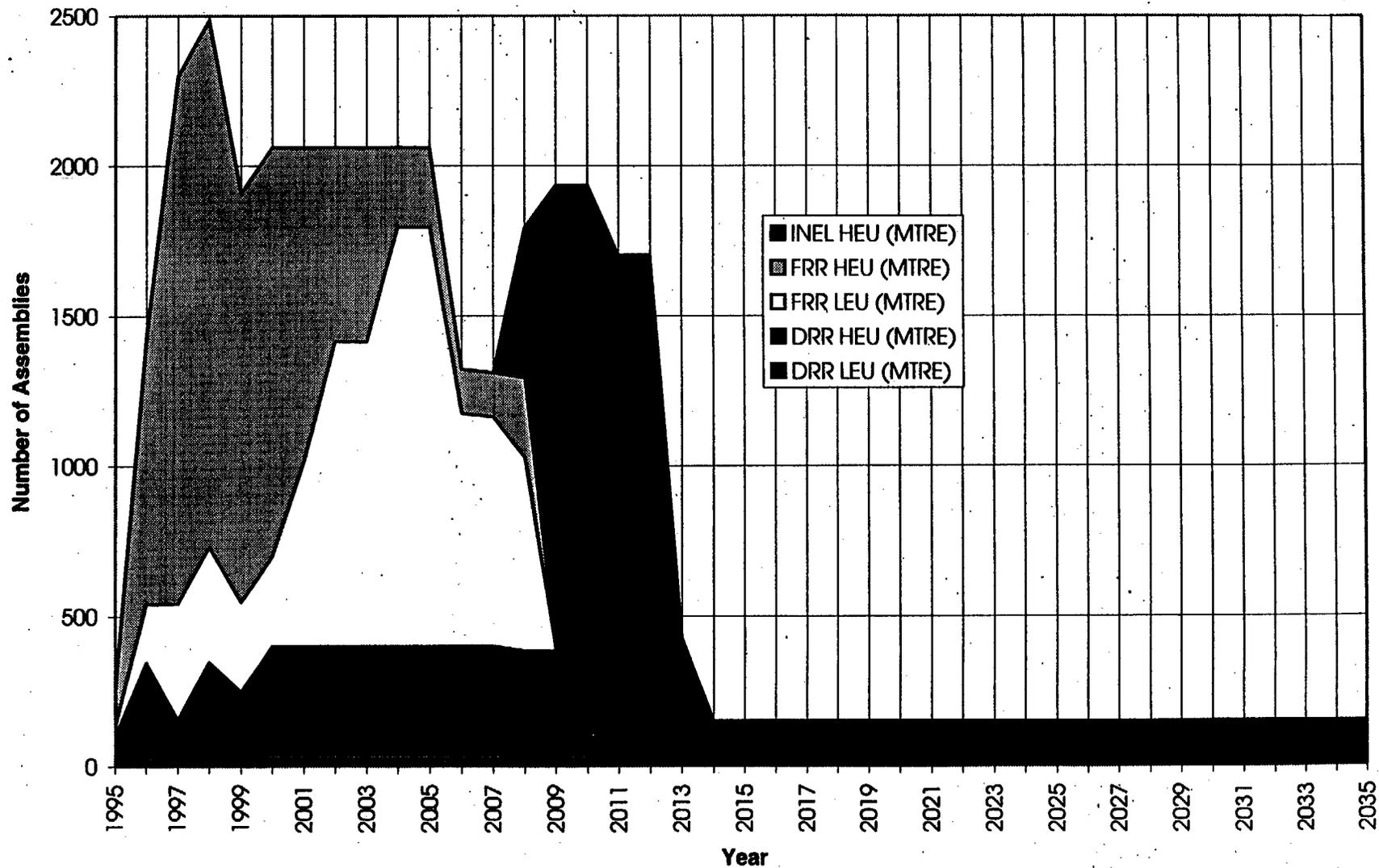


Figure C3.0-3 Assumed Non-MTRE SNF Assembly Deliveries to SRS

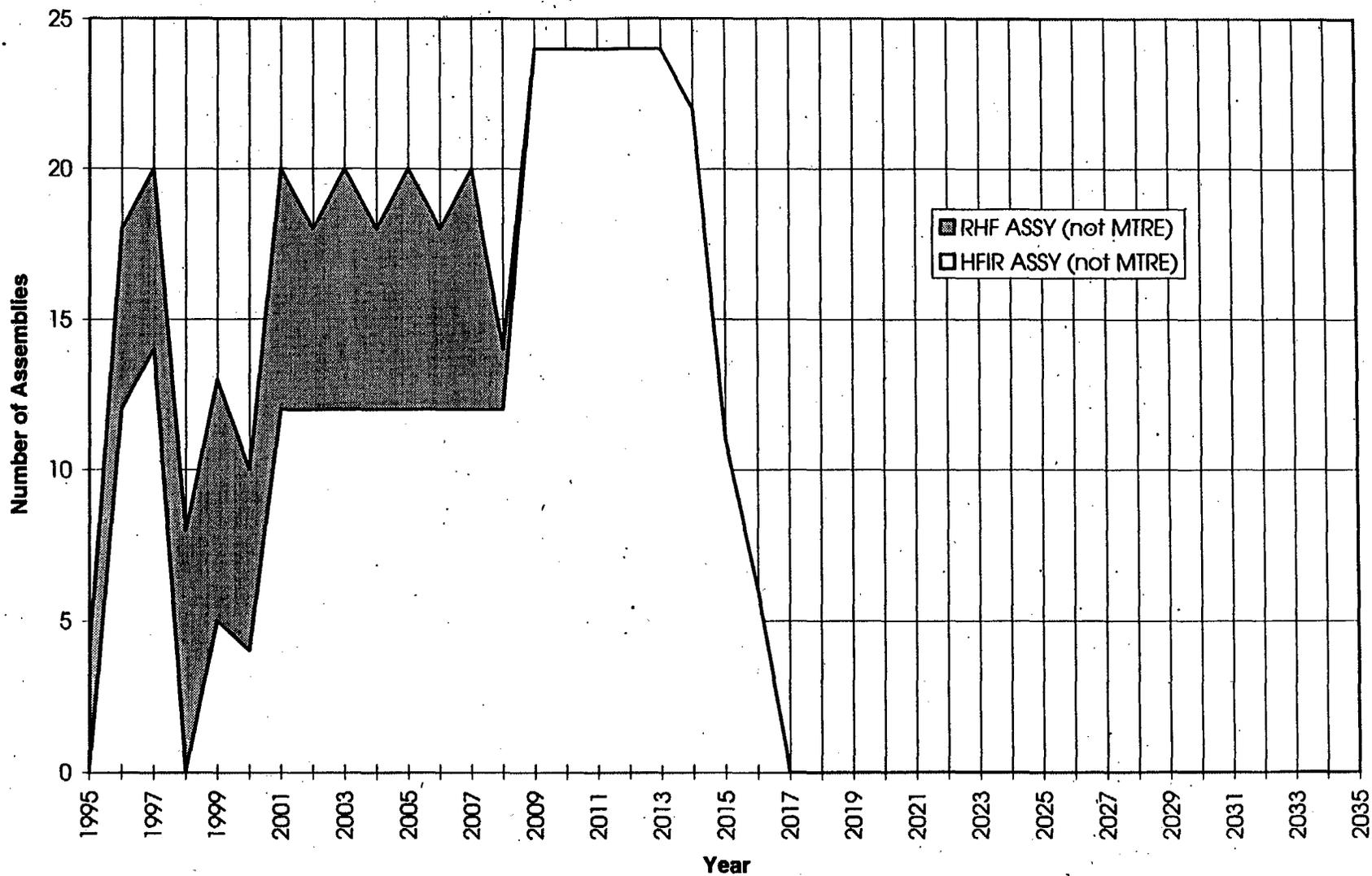


Table C3.0-3 Material Handling Assumptions

Parameter	Value
Off-site delivery casks and assemblies that can be handled in L-Basin	50%
HFIR / RHF Assemblies per CD Cask	5
MTRE Assemblies per CD Cask	52
Turnaround time for delivery cask (shifts)	10
Turnaround time for CD cask (shifts)	10
Total Available Time (Shifts per year)	1095
Facility Availability (%)	95%
Maximum Facility Availability (Shifts per year)	1040
Transfer Facility (EGG study) capacity (casks Delivered / year)	146

Table C3.0-4 Assumed Canister Quantities for Various Technologies

Technology	24" x 10' Direct Disposal	24" x 16' Direct Disposal	24" x 5' Direct Disposal	17" x 10' Co- Disposal	24" x 10' Glass Logs
Direct Disposal	80	370	650		
Co-Disposal				1400	
Press and Dilute (20%)				400	
Press and Dilute (2%)				1300	
Melt and Dilute				400	
Plasma Arc				400	
GMODS				800	
Dissolve and Vitrify				800	
Electrometallurgical					90 ²
Processing				300 ¹	120 ²

¹ Includes 300 canisters allocated for SNF delivered after end of canyon operations, assumed to be handled with direct disposal technology.

² Glass logs are assumed to be stored in the DWPF GWSB complex, which must be constructed for existing programs.

4.0 Definitions

For the purpose of evaluating costs, the following terminology was adopted:

- Research and Development and Demonstration Costs are those costs involved in developing a treatment technology, and demonstrating a high likelihood of success on a scale similar to that expected during production.
- Facility and Equipment Costs are those costs involved in the design and construction of new facilities for treatment or transfer and storage of SNF, including checkout and startup costs.
- Operations Costs are the costs involved in operating a new facility for its lifetime, including labor, materials, utilities, and maintenance contracts.
- Decontamination and Decommissioning Costs are the costs involved in preparing a facility for long-term unattended storage or use for unrestricted use.
- Wet Storage and Handling Costs are those costs involved in the operation of RBOF and L-basin as interim storage locations before a treatment or dry transfer and storage facility is available.
- Treatment Costs are those costs involved in changing the form of SNF to ensure its acceptability for storage, transportation, and disposal. These costs include the treatment module where each technology operates.
- Transfer and Packaging Costs are those costs involved with receiving shipping casks containing SNF, delivering it to a treatment or transfer function, and packaging the product for interim storage, transportation and disposal. These are common costs for all technologies.
- Interim Storage Costs are costs involved in storing the products of the treatment technology until the repository is operational and ready to receive this material.
- Repository Fee is the fixed-cost portion of repository costs, and is assumed to be an equitable share of the total repository cost, proportional to the volume of research reactor SNF compared to all SNF in the repository.
- Repository Operations Cost is the variable-cost portion of repository costs, assumed to be proportional to the number of waste packages, at a unit rate determined by the waste package configuration.
- Transportation Cost in this report is that portion of the cost of the operation of the rail transportation system for delivery of research reactor SNF material from SRS to the repository. The cost of delivering this material to SRS is the same for all treatment options.

5.0 Summary of Results

Table C5.0-1 shows the total comparative life cycle cost, and the discounted comparative total life cycle cost for each option. Table C5.0-2 presents detailed summary comparative cost information for the various treatment technologies. These comparative costs are subdivided into those due to existing wet storage facilities, transfer and packaging facilities, treatment facilities, interim storage facilities, repository costs, and transportation costs, and cost adjustments.

Table C5.0-1 Net Present Value of System Comparative Costs

Technology	Total System Cost (\$1996)	Net Present Value System Cost (1996)
Direct Disposal	\$1,362	\$888
Co-Disposal	\$1,176	\$832
Press and Dilute (20%)	\$1,395	\$979
Press and Dilute (2%)	\$1,556	\$1,058
Melt and Dilute	\$1,348	\$963
Plasma Arc	\$1,918	\$1,326
GMODS	\$1,902	\$1,298
Dissolve and Vitrify	\$1,989	\$1,341
Electrometallurgical	\$1,589	\$1,110
Processing (Baseline)	\$1,231	\$763

These Comparative costs include relative uncertainty adjustments for transfer, treatment and storage costs that reflect the potential for increased costs for the less mature technologies. The team has also developed an estimate range within which a more complete and better developed cost projection based on more design detail would likely fall. Figure C5.0-1 shows the uncertainty ranges associated with each treatment technology. The relative uncertainty adjustments and cost ranges are presented in Section 9.0.

Figures C5.0-2 through C5.0-5 show the cumulative system expenditures for the various treatment technologies. The net present worth of the cash flows for each technology is calculated at 3% and presented in Tables 4.2-2 and C5.0-1.

The objective of the cost evaluation was to consistently compare costs among the technologies using available pre-conceptual design information. The approach used to develop the estimate and apply uncertainties reflects the application of considerable experience and judgement by the Team.

TABLE C5.0-2 SUMMARY OF COMPARISON COST INFORMATION (Millions of 1996 Dollars¹)

Option	Wet Storage and Handling	Transfer and Packaging	Treatment	Interim Storage	Disposal			Adjustments ²	Total
					Transportation	Repository Fee	Repository Operations		
Direct Disposal (1100 Small Packages)	282	440	0	144	37	30	368	61	1362
Direct Disposal - CoDisposal (1400 Co-Disposal Pkgs)	282	469	0	151	42	30	140	61	1176
Press and Dilute (400 Co-Disposal Pkgs - 20%)	347	459	323	116	19	30	40	61	1395
Press and Dilute (1300 Co-Disposal Pkgs - 2%)	347	479	323	146	40	30	130	61	1556
Melt and Dilute (400 Co-Disposal Pkgs)	347	432	383	97	19	30	40	0	1348
Plasma Arc (400 Co-Disposal Pkgs)	456	418	861	93	19	30	40	0	1918
GMODS (800 Co-Disposal Pkgs)	456	427	774	106	28	30	80	0	1902
Dissolve and Vitrify (800 Co-Disposal Pkgs)	456	427	861	106	28	30	80	0	1989
Electrometallurgical (90 Glass Logs)	436	401	954	3	14	30	11	-223	1625
Processing / Direct Disposal (120 Glass Logs, 300 Co-Disposal Pkgs)	435	173	643	60	22	30	44	-175	1231

Note 1: The total cost on which the comparative evaluation of options is based.

Note 2: Adjustments include charges for a conditioning (hot vacuum drying) facility for fuels which require that treatment for direct disposal. If instead, these fuels are processed, the adjustment is reduced to \$30 M. Also includes credits for sale of U-235 when appropriate.

Figure C5.0-1 Range of Relative Costs

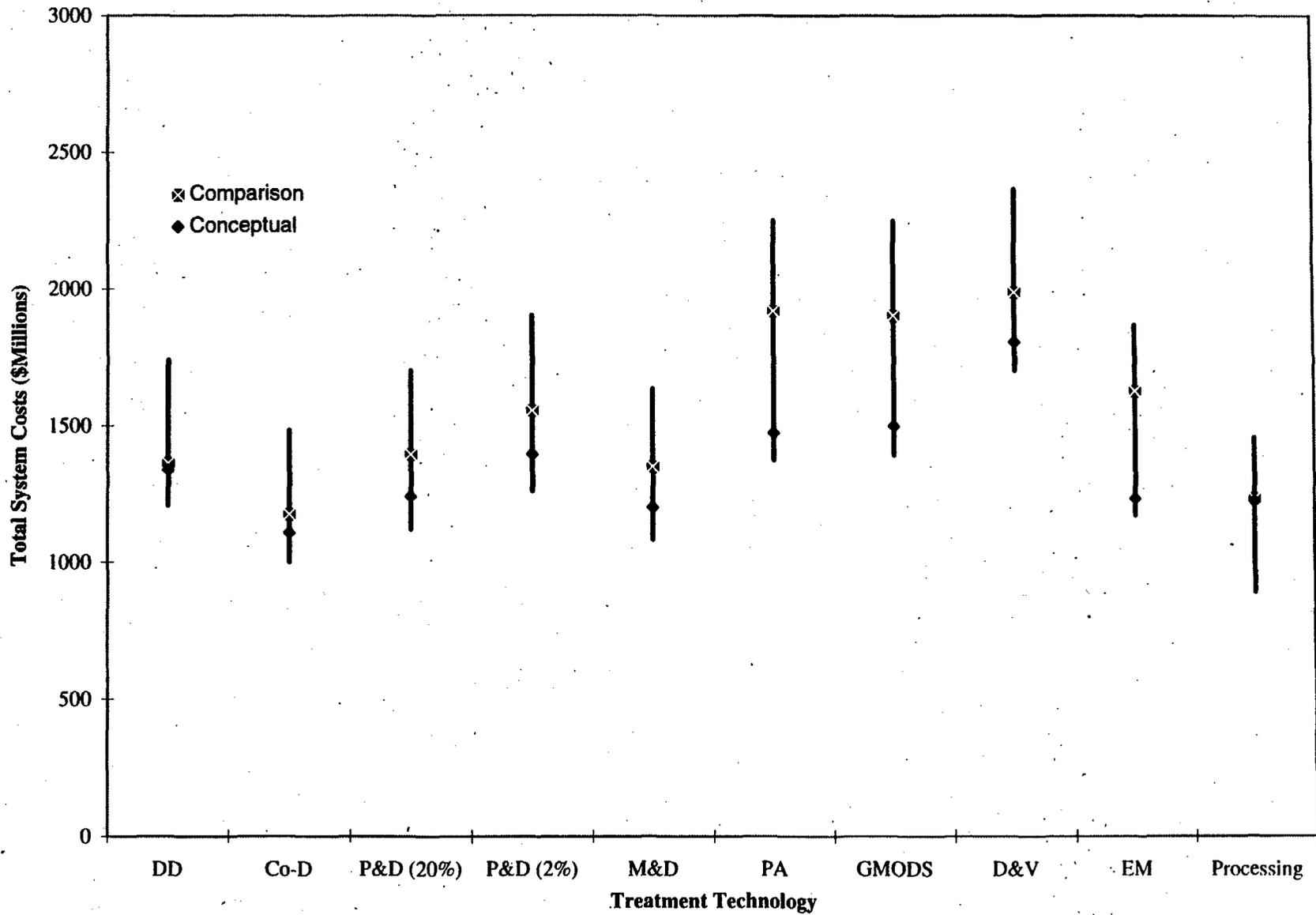


Figure C5.0-2 Cumulative Expenditures for Direct Disposal Methods

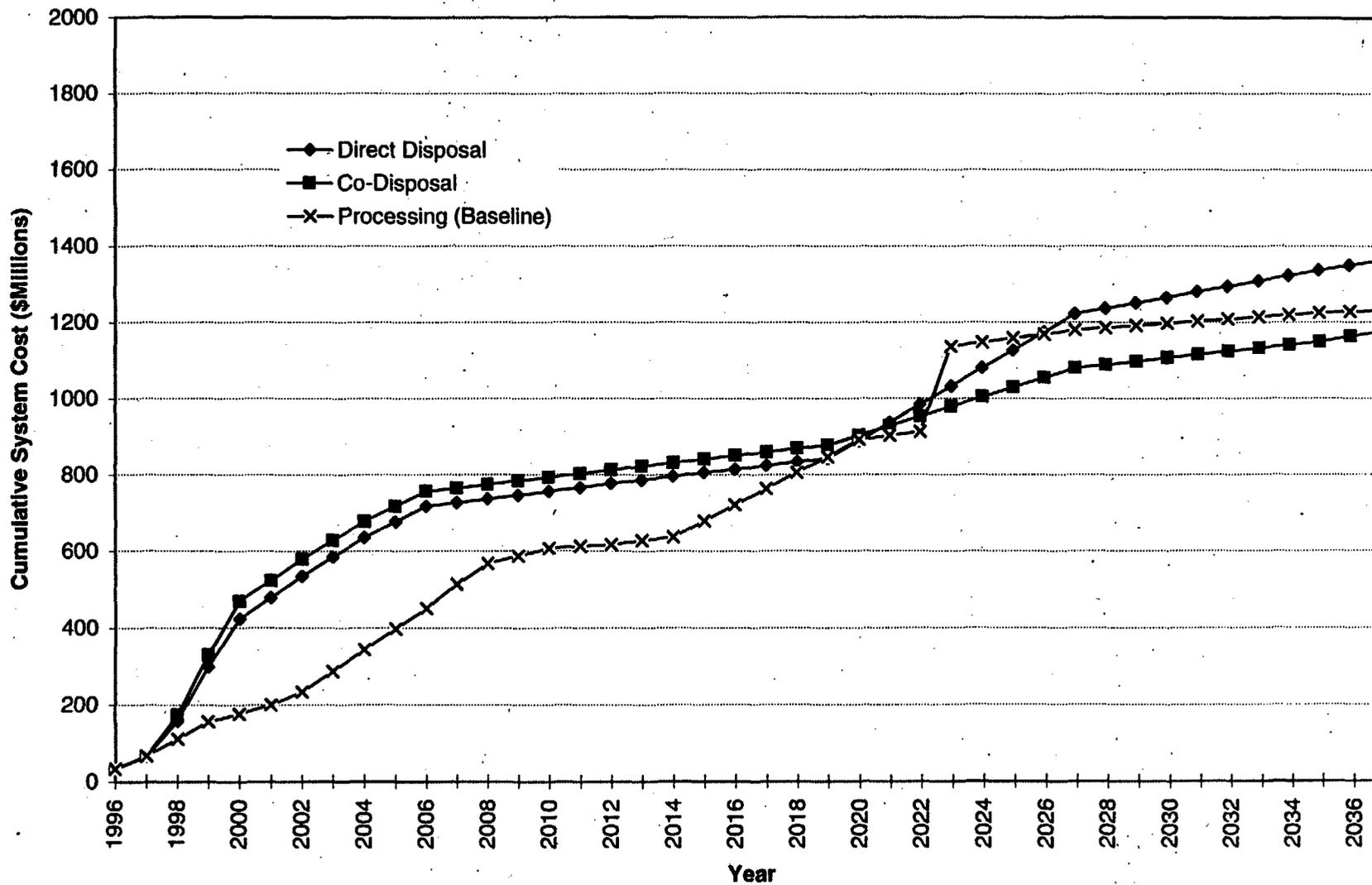


Figure C5.0-3 Cumulative Expenditures for HEU Dilution Methods

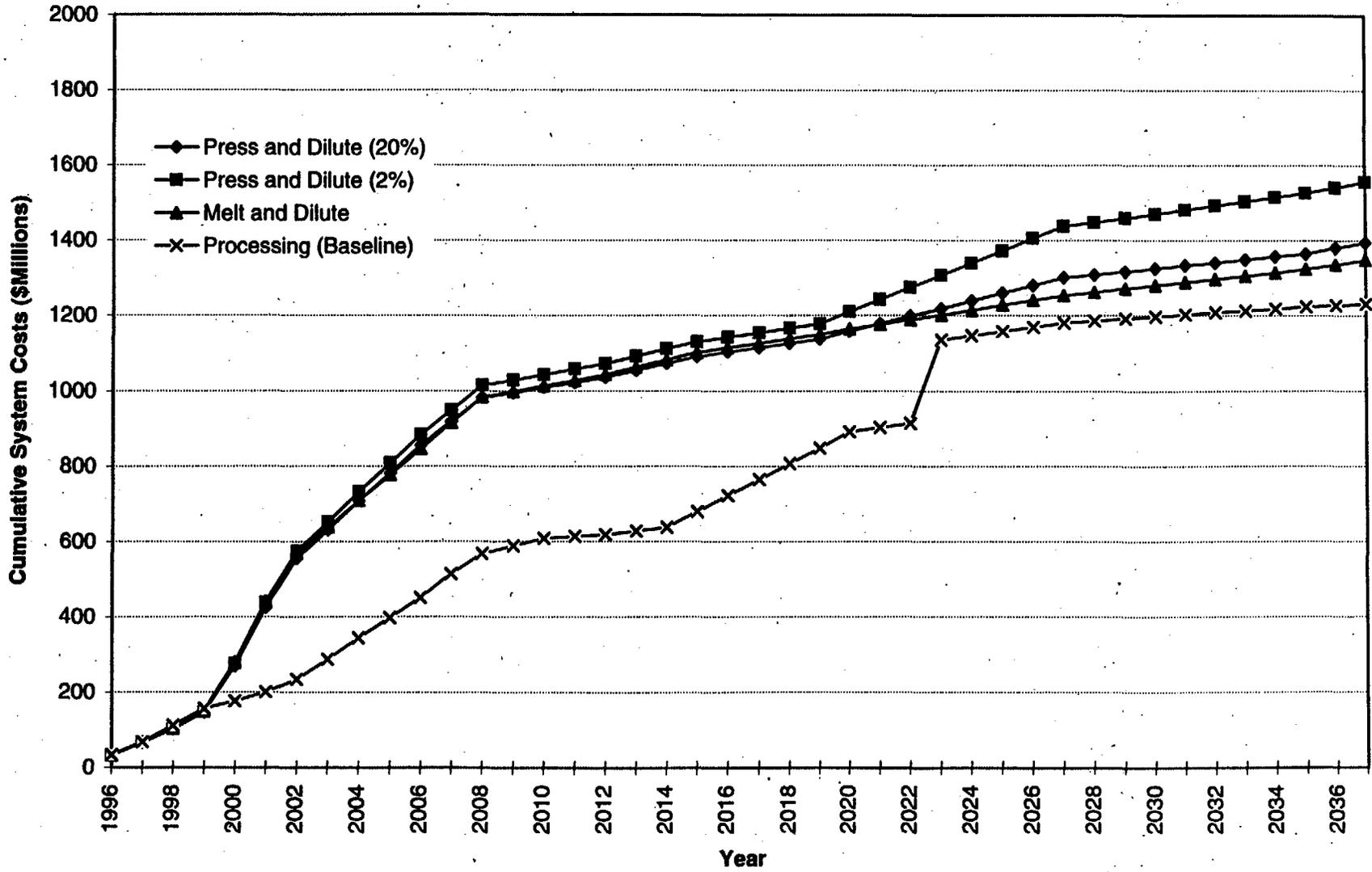


Figure C5.0-4. Cumulative Expenditure for Advanced Treatment Methods

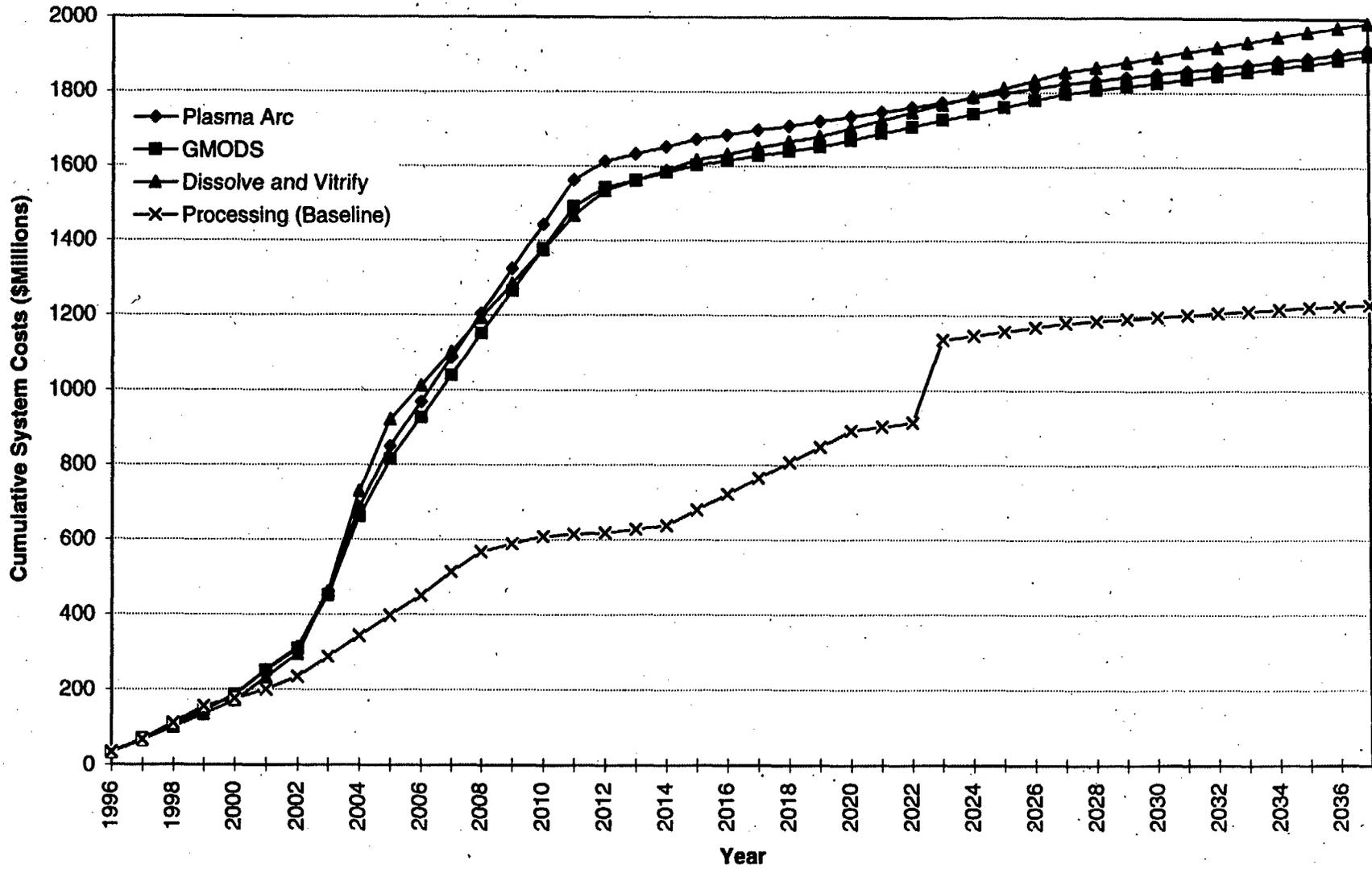
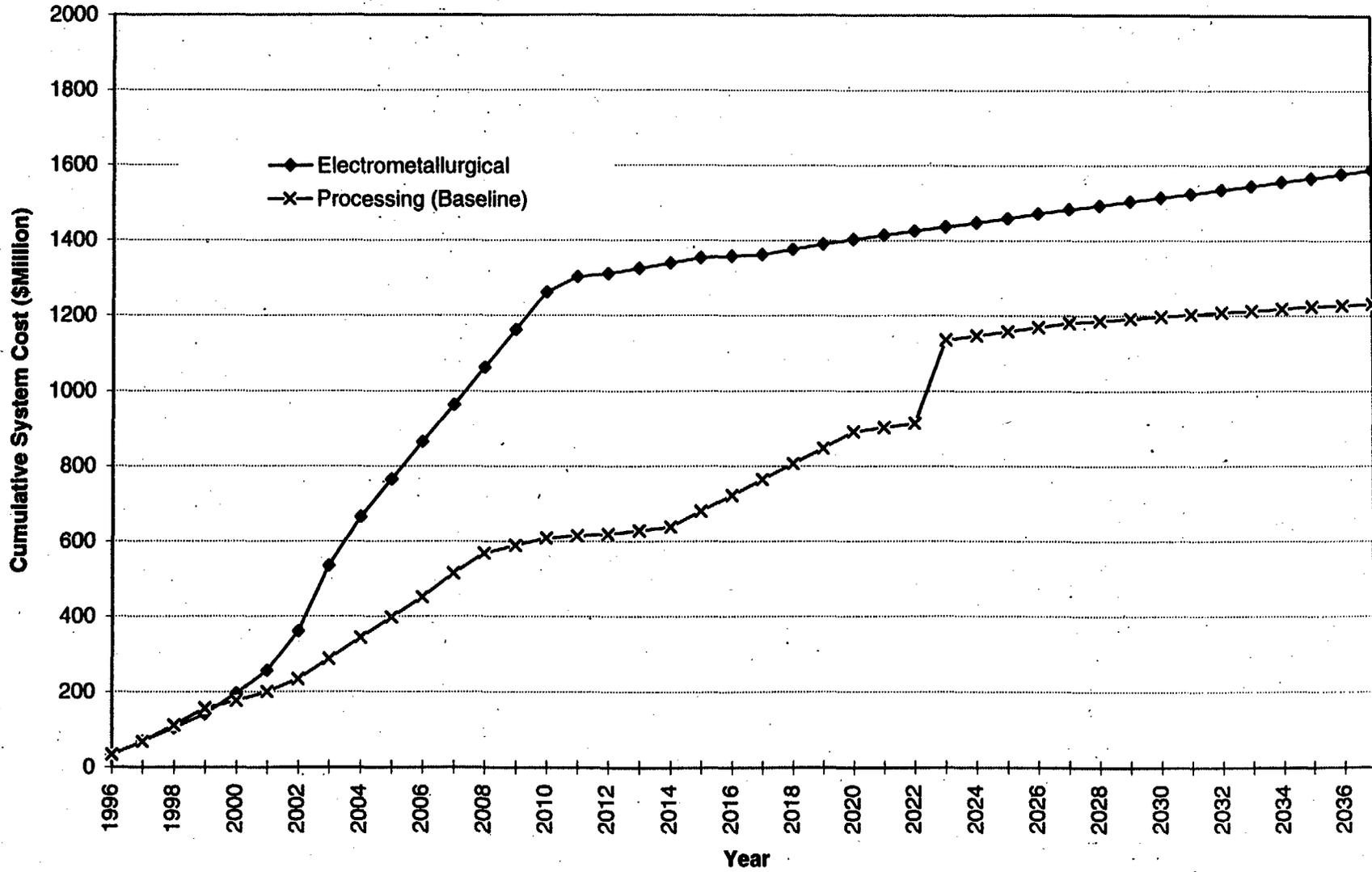


Figure C5.0-5 Cumulative Expenditures for Uranium Separation and Dilution Method



6.0 System Operations and Schedule

For all treatment technologies evaluated, existing SRS facilities were integrated into a plan with proposed new facilities. Until a new treatment technology is on-line, off-site shipments will continue to be received at RBOF and L-basin. RBOF storage is projected to be filled by the end of 1996, after which time new rack storage for approximately 16,000 SNF assemblies in L-basin will be available. After 1996, RBOF can only serve as a wet transfer point for handling off-site casks which are too tall for handling in L-basin's shallower pool transfer area. Approximately half the cask deliveries from off-site are directed to RBOF, where the SNF assemblies are transferred to high-capacity on-site CD casks for trans-shipment to L-basin. Until the new treatment technology is available, the inventory of SNF stored in the basins continues to grow. If no new treatment facilities are available by 2003, additional storage will need to be installed in L-basin. Table C6.0-1 shows the peak inventory of SNF assemblies projected for L-basin based on different dates for start of operations at a new treatment facility. Figure C6.0-1 shows the overall operational flow assumed for the site.

Table C6.0-1 Projected Peak Inventory of L-basin Storage

Start Date of New Treatment Facility Operations	L-Basin Inventory (MTRE assemblies)	L-Basin Inventory (non-MTRE assemblies)
2001	10,200	70
2003	14,300	110
2005	18,500	150
2006	20,500	170

Different operational start dates were assumed for each treatment technology, depending on its relative maturity. The assumed startup dates are shown below:

- 2001 startup for Direct Disposal, Co-Disposal
- 2003 startup for Press and Dilute, Melt and Dilute
- 2005 startup for Electrometallurgical
- 2006 startup for Plasma Arc, GMODS, Dissolve and Vitrify

It is recognized that all these start dates are extremely aggressive, and assume that traditional appropriations, budgeting and management practices will be significantly compressed to achieve an early solution.

With the assumed schedule of cask and assembly deliveries shown in Tables C3.0-1 and C3.0-2, an analysis was made of site operations both before and after initial operation of the new treatment facility. Tables C6.0-2 and C6.0-3 show the assumed receipt and trans-shipment of SNF assemblies and casks at the wet storage basins for a technology that begins operation in 2001. Tables C6.0-4 and C6.0-5 show the same information for the transfer area at the new

treatment facility. Similar tables were also developed for the other start dates. The site operations have been selected considering assumptions about the required handling times for receiving and trans-shipping casks at existing and future facilities. The limiting factor for emptying the wet basin storage is found to be the receiving capacity at the new transfer facility. As a result, it will take six or seven years to empty the wet basins after the new technology begins operation, depending on how much inventory has been accumulated in L-basin. (Trade-off studies should be performed during detailed design to determine whether it would be cost-effective to increase the new transfer capacity to empty the basins sooner.)

After treatment operations begin, it is assumed that off-site SNF receipts would be sent directly to the transfer area described in Section 3.2, and that SNF would be transferred from wet storage basins as quickly as the treatment capacity permits. The objective is to de-inventory the wet storage basins within six or seven years to minimize operating costs. Once the basins are emptied and decommissioned, a significant annual throughput still remains until the last INEL SNF is received around 2012. Operations at the new facility can then be scaled back for a few years until the repository begins operation.

It is assumed that all inventory from the interim storage facility will be shipped to the repository over an eight-year period beginning in 2020; shipments after that will be made as required to keep up with incoming receipts from operating reactors.

Treatment technologies which do not change the chemical composition of the SNF are assumed to vent the packages before shipment to the repository. This assumption requires that the packages be returned to the transfer facility. Products from other technologies that have either melted or vitrified the material can be directly shipped from the storage facility without the need for venting, saving operational expense.

Using these analyses and assumptions as inputs, overall project schedules were developed for each of the treatment options. Figures C6.0-1 through C6.0-4 show the projected facility utilizations for treatment startup in 2001, 2003, 2005, and 2006. Figure C6.0-5 shows a schedule of major phases of design, construction, and operation for direct disposal and co-disposal. Figure C6.0-6 shows a similar schedule for the press and dilute option, and Figure C6.0-7 shows the schedule for the melt and dilute option. While these schedules are not identical to those in Volume I, the differences are too small to make a significant difference. These two figures differ in the requirement for a hot vacuum drying facility to address Uranium metal SNF, and in the requirement to vent press and dilute packages before shipment to the repository. Figures C6.0-8 and C6.0-9 show system schedules for those technologies assumed to begin operations in 2005 and 2006, respectively.

Table C6.0-2 Throughput of SNF Assemblies at Existing Basins
(2001 Transfer start)

FISCAL YEAR	OFF-SITE DELIVERIES TO SRS		RBOF OPERATIONS								L-BASIN OPERATIONS							
			MTRE				HFIR/RHF				MTRE				HFIR/RHF			
	MTRE	HFIR & RHF	FROM OFF-SITE	TO L-BASIN (CD CASK)	TO DRY TRANS FACIL (CD CASK)	CUM RBOF INVENT	FROM OFF-SITE	TO L-BASIN (CD CASK)	TO DRY TRANS FACIL (CD CASK)	CUM RBOF INVENT	FROM OFF-SITE	FROM RBOF (CD CASK)	TO DRY TRANS FACIL (CD CASK)	CUM L-BASIN INVENT	FROM OFF-SITE	FROM RBOF (CD CASK)	TO DRY TRANS FACIL (CD CASK)	CUM L-BASIN INVENT
1995	252	4	252			3004	4			4								0
1996	1431	18	716	716		3004	9	9		4	716	716		1431	9	9		18
1997	2299	20	1150	1150		3004	10	10		4	1150	1150		3730	10	10		38
1998	2486	8	1243	1243		3004	4	4		4	1243	1243		6216	4	4		46
1999	1909	13	955	955		3004	7	7		4	955	955		8125	7	7		59
2000	2063	10	1032	1032		3004	5	5		4	1032	1032		10188	5	5		69
2001	2061	20			500	2504				4			1900	8288			12	57
2002	2063	18			500	2004				4			1900	6388			12	45
2003	2061	20			500	1504				4			1900	4488			12	33
2004	2063	18			500	1004				4			1900	2588			12	21
2005	2061	20			500	504				4			1900	688			12	9
2006	1323	18			504	0			4	0			688	0			9	0

Table C6.0-3 Throughput of Transportation Casks at Existing Basins
(2001 Transfer Start)

FISCAL YEAR	OFF-SITE CASK DELIVERIES TO SRS		RBOF OPERATIONS						L-BASIN OPERATIONS					
			MTRE			HFIR/RHF			MTRE			HFIR/RHF		
	MTRE	HFIR & RHF	FROM OFF SITE	TO L-BASIN (CD CASK)	TO DRY TRANS FACIL (CD CASK)	FROM OFF SITE	TO L-BASIN (CD CASK)	TO DRY TRANS FACIL (CD CASK)	FROM OFF SITE	FROM RBOF (CD CASK)	TO DRY TRANS FACIL (CD CASK)	FROM OFF SITE	FROM RBOF (CD CASK)	TO DRY TRANS FACIL (CD CASK)
1995	11	0	11			0								
1996	47	12	24	14		6	2		24	14		6	2	
1997	76	14	38	22		7	2		38	22		7	2	
1998	81	4	41	24		2	1		41	24		2	1	
1999	73	5	37	18		3	1		37	18		3	1	
2000	85	4	43	20		2	1		43	20		2	1	
2001	85	12			10			0			37			2
2002	85	12			10			0			37			2
2003	85	12			10			0			37			2
2004	85	12			10			0			37			2
2005	85	12			10			0			37			2
2006	60	12			10			1			13			2

**Table C6.0-4 Throughput of SNF Assemblies at New Treatment Facilities
(2001 Transfer start)**

FISCAL YEAR	OFF-SITE DELIVERIES TO SRS		TRANSFER FACILITY OPERATIONS							
	MTRE	HFIR & RHF	MTRE				HFIR/RHF			
			FROM OFF-SITE	FROM RBOF (CD CASK)	FROM L-BASIN (CD CASK)	TOTAL	FROM OFF-SITE	FROM RBOF (CD CASK)	FROM L-BASIN (CD CASK)	TOTAL
1995	252	4								
1996	1431	18								
1997	2299	20								
1998	2486	8								
1999	1909	13								
2000	2063	10								
2001	2061	20	2061	500	1900	4461	20	0	12	32
2002	2063	18	2063	500	1900	4463	18	0	12	30
2003	2061	20	2061	500	1900	4461	20	0	12	32
2004	2063	18	2063	500	1900	4463	18	0	12	30
2005	2061	20	2061	500	1900	4461	20	0	12	32
2006	1323	18	1323	504	688	2515	18	4	9	31
2007	1311	20	1311			1311	20			20
2008	1795	14	1795			1795	14			14
2009	1936	24	1936			1936	24			24
2010	1935	24	1935			1935	24			24
2011	1702	24	1702			1702	24			24
2012	1702	24	1702			1702	24			24
2013	426	24	426			426	24			24
2014	150	22	150			150	22			22
2015	150	11	150			150	11			11
2016	150	6	150			150	6			6
2017	150	0	150			150				
2018	150	0	150			150				
2019	150	0	150			150				
2020	150	0	150			150				
2021	150	0	150			150				
2022	150	0	150			150				
2023	150	0	150			150				
2024	150	0	150			150				
2025	150	0	150			150				
2026	150	0	150			150				
2027	150	0	150			150				
2028	150	0	150			150				
2029	150	0	150			150				
2030	150	0	150			150				
2031	150	0	150			150				
2032	150	0	150			150				
2033	150	0	150			150				
2034	150	0	150			150				
2035	150	0	150			150				
TOTAL	36179	380								

Table C6.0-5 Throughput of Transportation Casks at New Treatment Facilities.
(2001 Transfer Start)

FISCAL YEAR	OFF-SITE CASK DELIVERIES TO SRS		TRANSFER FACILITY OPERATIONS					
			MTRE			HFIR/RHF		
	MTRE	HFIR & RHF	FROM OFF-SITE	FROM RBOF (CD CASK)	FROM L-BASIN (CD CASK)	FROM OFF-SITE	FROM RBOF (CD CASK)	FROM L-BASIN (CD CASK)
1995	11	0						
1996	47	12						
1997	76	14						
1998	81	4						
1999	73	5						
2000	85	4						
2001	85	12	85	10	37	12	0	2
2002	85	12	85	10	37	12	0	2
2003	85	12	85	10	37	12	0	2
2004	85	12	85	10	37	12	0	2
2005	85	12	85	10	37	12	0	2
2006	60	12	60	10	13	12	1	2
2007	59	12	59			12		
2008	69	12	69			12		
2009	58	24	58			24		
2010	58	24	58			24		
2011	44	24	44			24		
2012	44	24	44			24		
2013	16	24	16			24		
2014	10	22	10			22		
2015	10	11	10			11		
2016	10	6	10			6		
2017	10	0	10			0		
2018	10	0	10			0		
2019	10	0	10			0		
2020	10	0	10			0		
2021	10	0	10			0		
2022	10	0	10			0		
2023	10	0	10			0		
2024	10	0	10			0		
2025	10	0	10			0		
2026	10	0	10			0		
2027	10	0	10			0		
2028	10	0	10			0		
2029	10	0	10			0		
2030	10	0	10			0		
2031	10	0	10			0		
2032	10	0	10			0		
2033	10	0	10			0		
2034	10	0	10			0		
2035	10	0	10			0		
TOTAL	1426							

Figure 6.0-1 Projected Facility Utilization - 2001 Startup

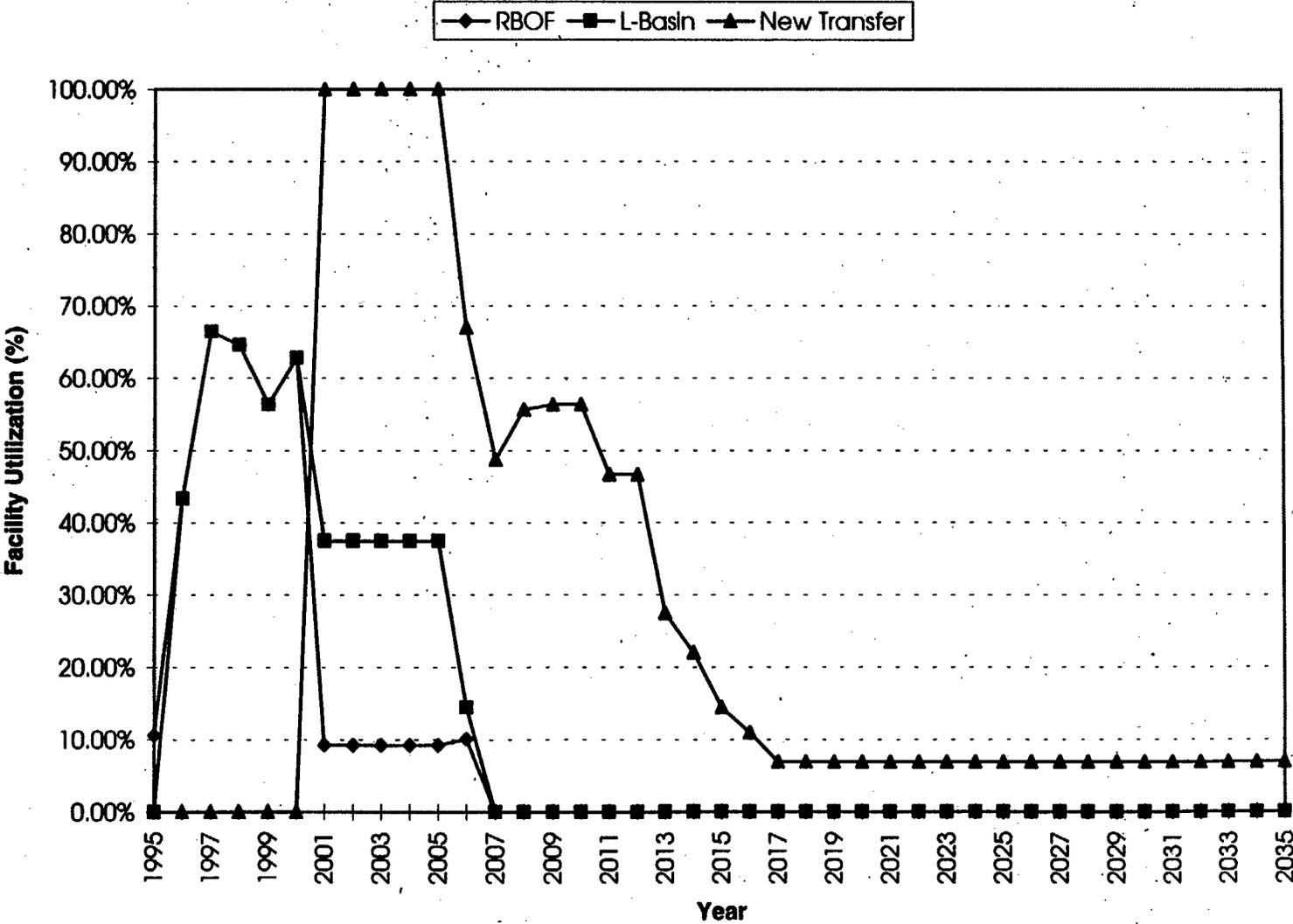


Figure 6.0-2 Projected Facility Utilization - 2003 Startup

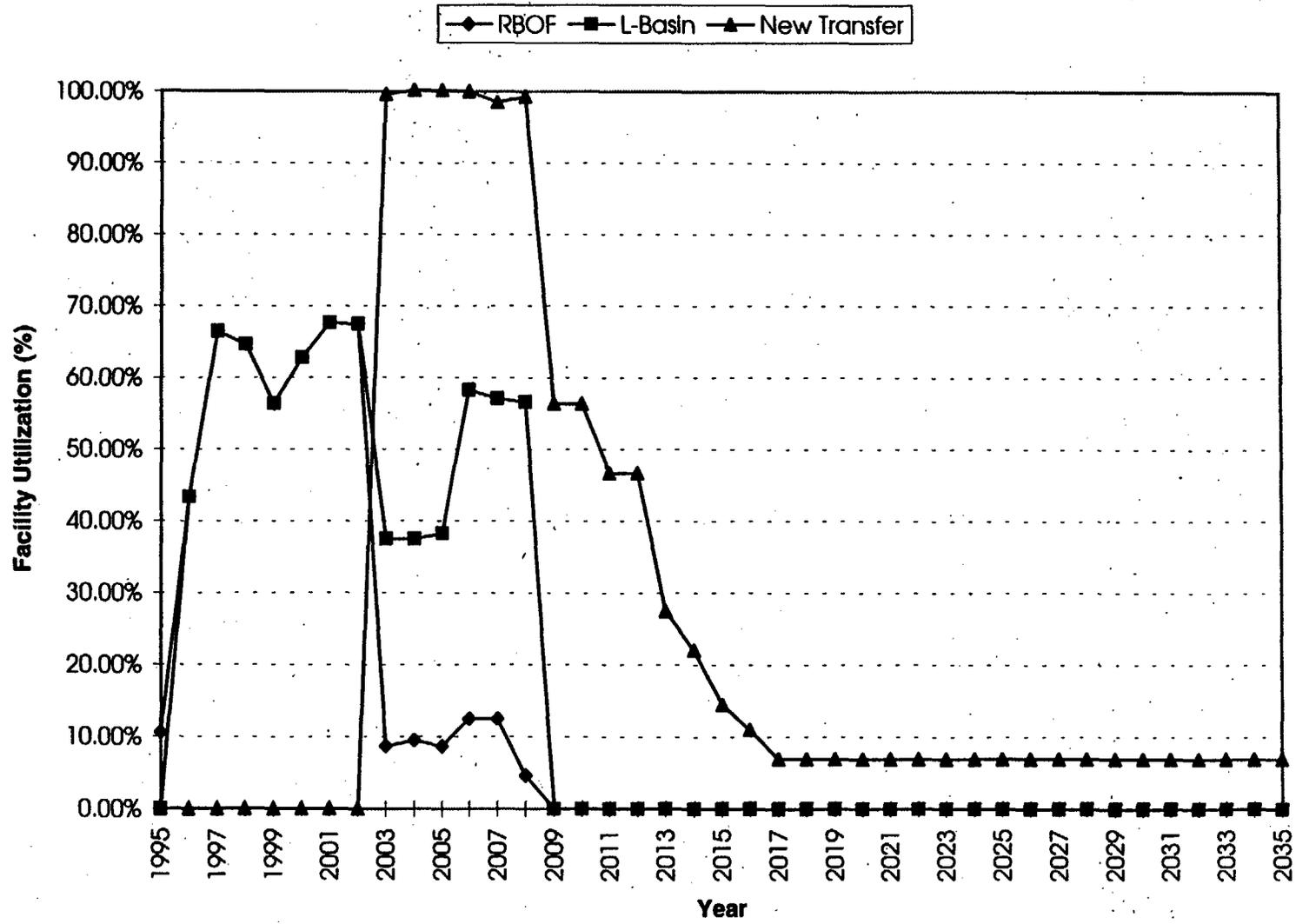


Figure 6.0-3 Projected Facility Utilization - 2005 Startup

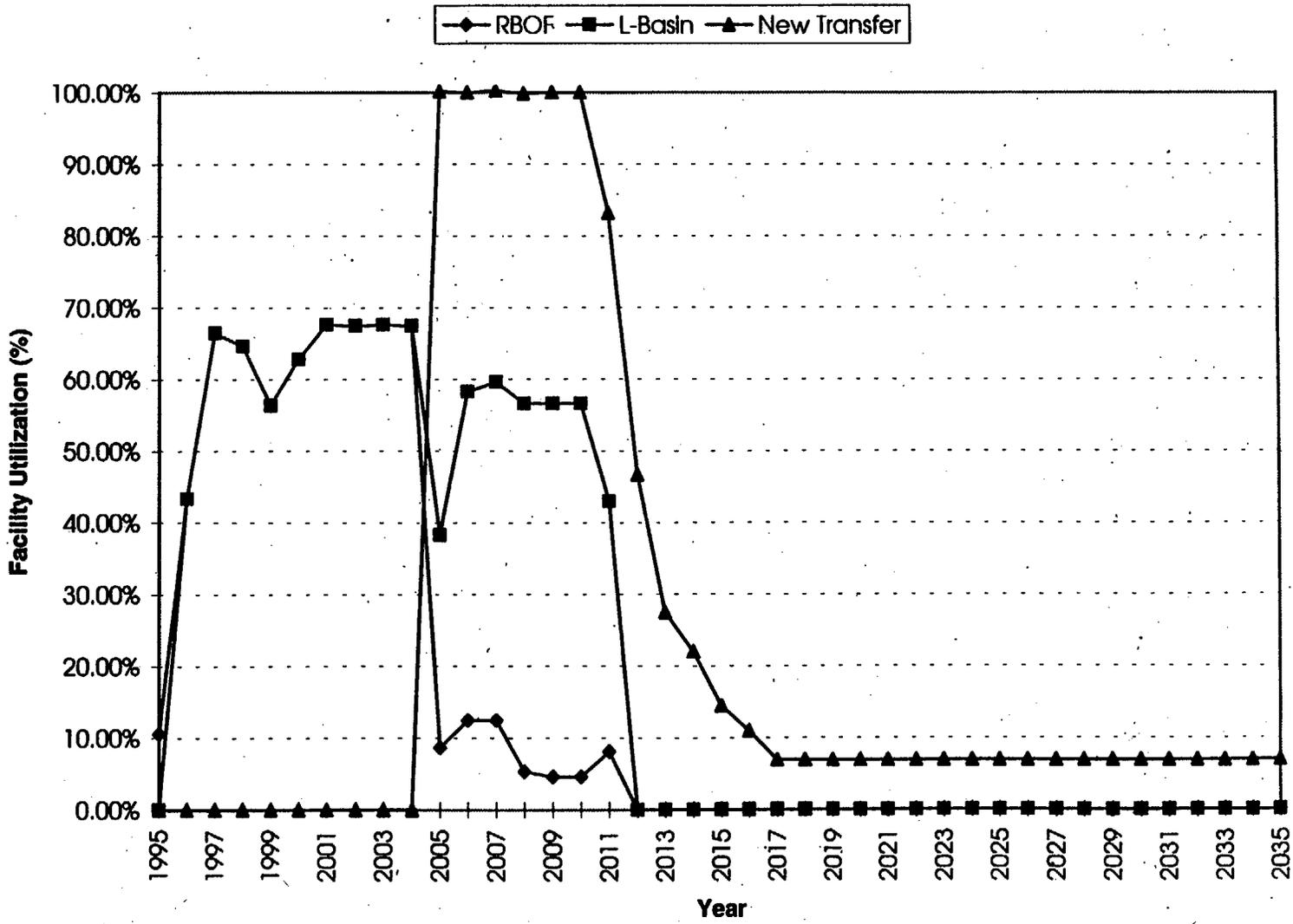


Figure 6.0-4 Projected Facility Utilization - 2006 Startup

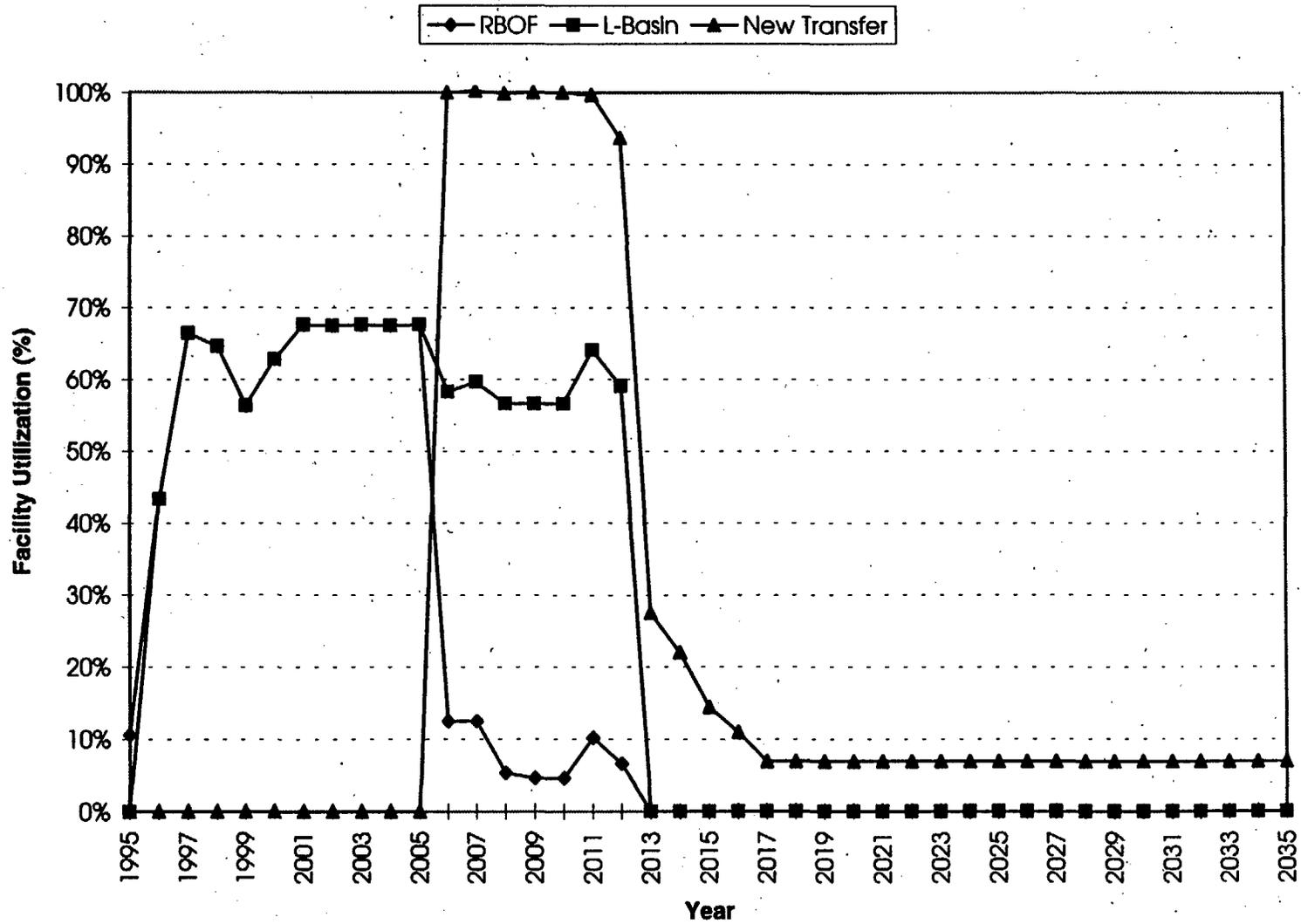


Figure C6.0-5 Overall Schedule for Direct Disposal Methods

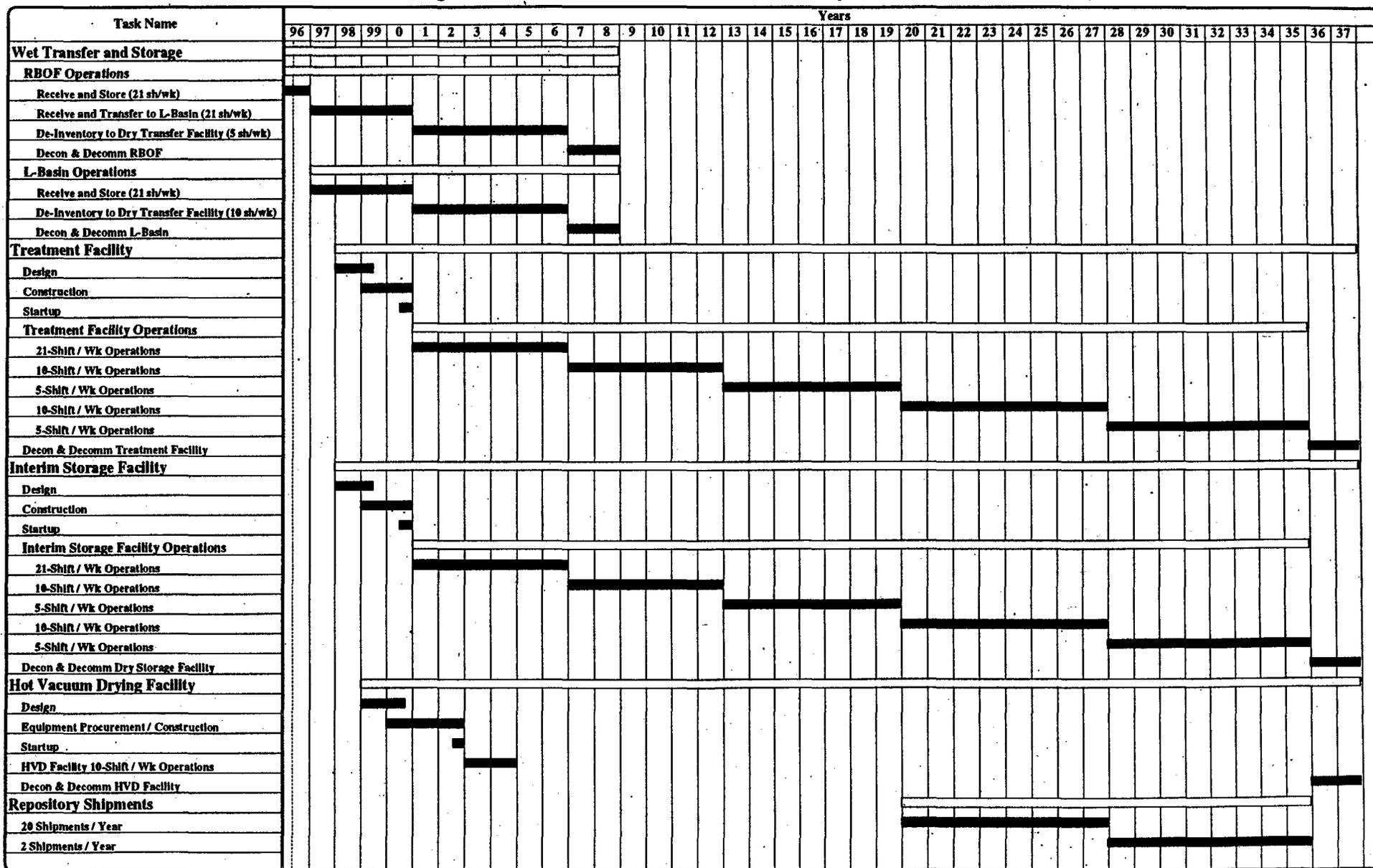


Figure C6.0-6 Overall Schedule for Press and Dilute Option

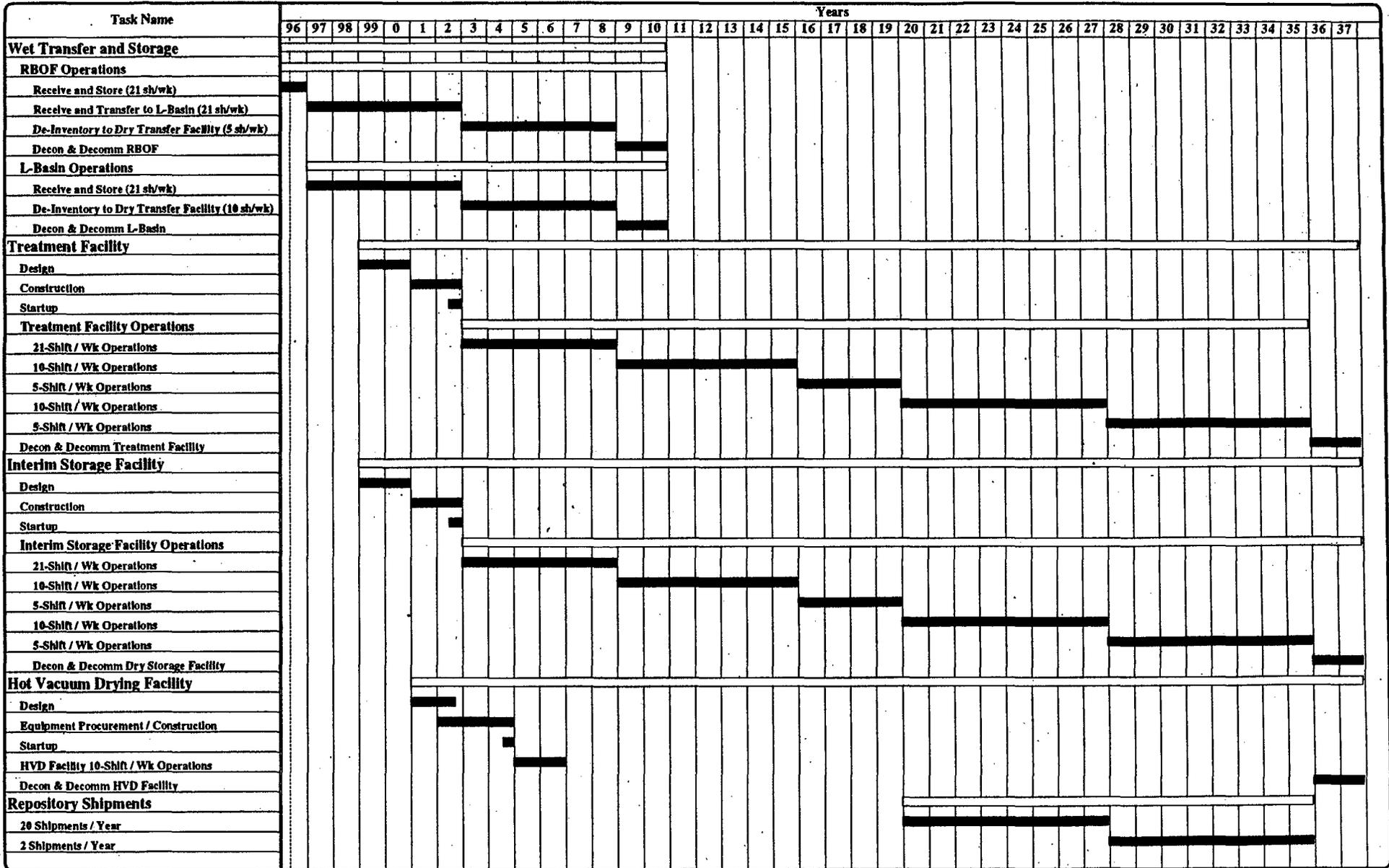


Figure C6.0-7 Overall Schedule for Melt and Dilute Option

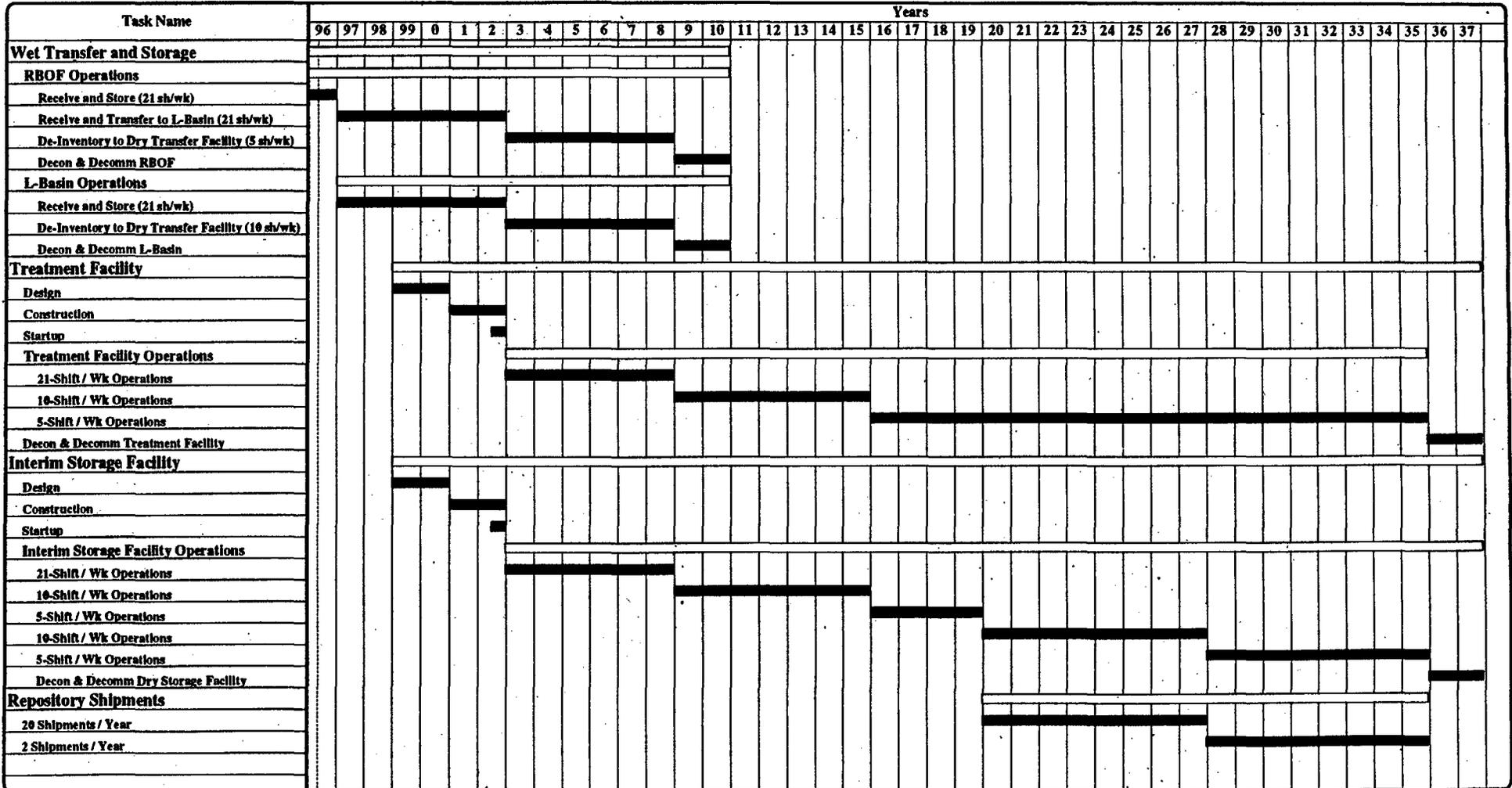


Figure C6.0-8 Overall Schedule for Electrometallurgical Option

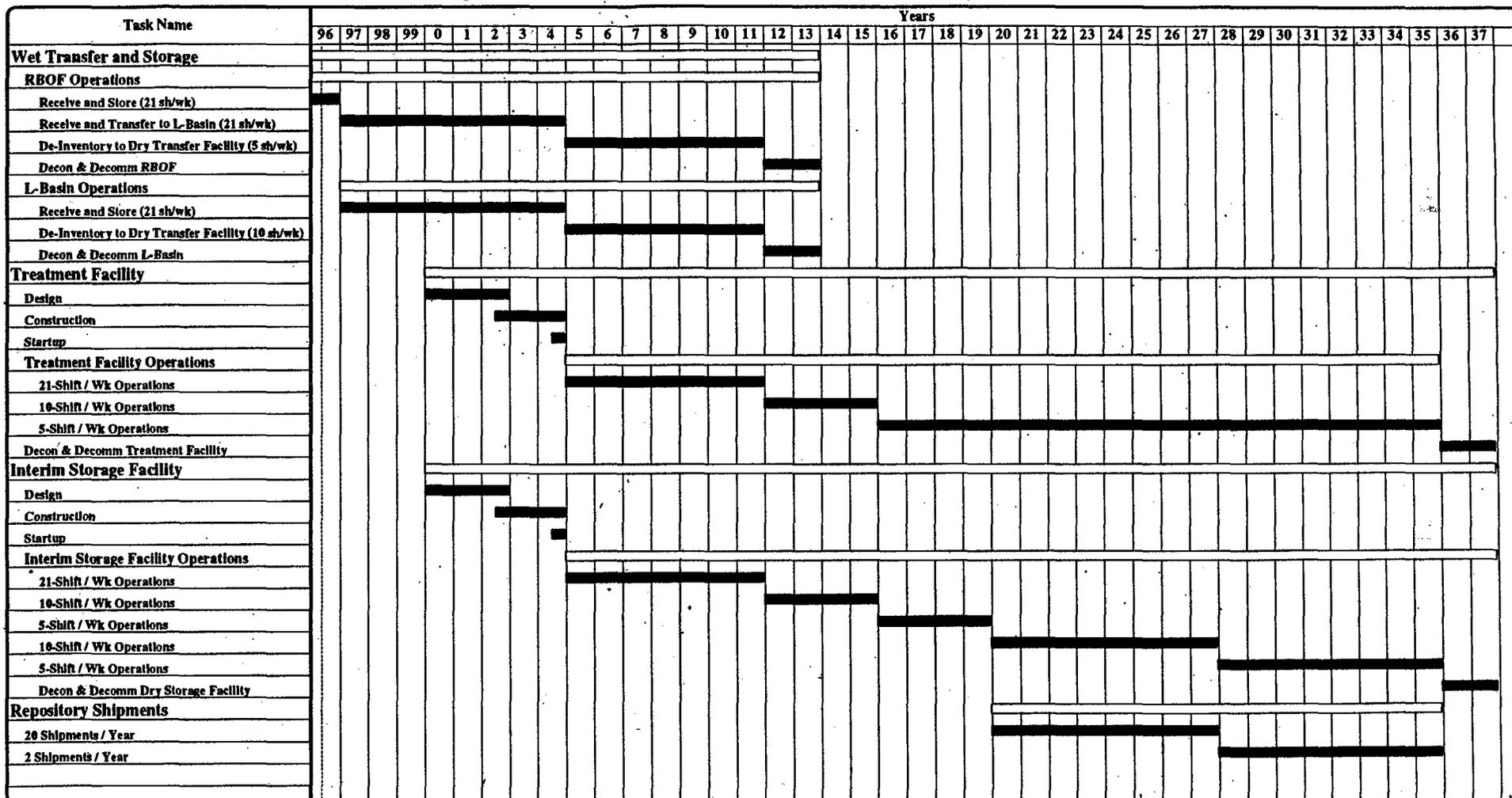
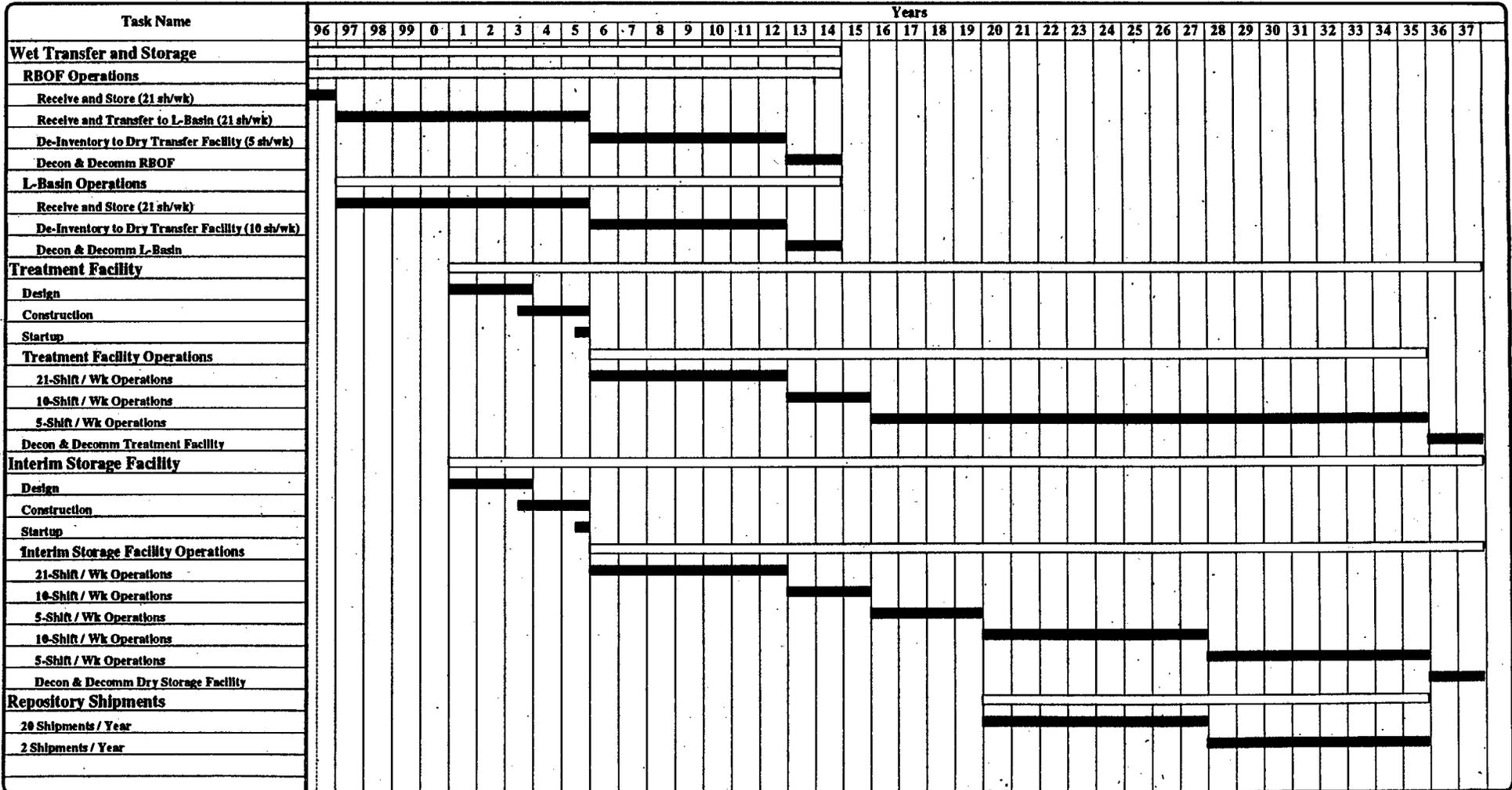


Figure C6.0-9 Overall Schedule for Advanced Treatment Methods



7.0 General Conceptual Costs

This section presents cost estimate information for items which are common to all technologies. Some costs are not specifically estimated for this study, such as safeguards and security costs, and health physics operations costs. Since inputs such as safeguards and security implementing requirements may be different for HEU compared to LEU, this issue should be studied in greater detail once the field of technologies is narrowed. For the purposes of this study, such costs are assumed to be the same for all alternatives, and will therefore not affect the selection process.

7.1 Wet Storage Costs

All options initially receive SNF into wet lag storage at RBOF and L-basin before the treatment or transfer facilities are available. The estimated costs for operating these facilities are based on historical data, as described in WSRC-RP-95-798, Savannah River Site Evaluation of Spent Nuclear Fuel Options. The operating cost for RBOF and L-basin is \$15 million per year, and \$17.2 million per year, respectively, on a 21-shifts per week basis. For periods of less intense activity at these facilities, these figures are scaled back to 5- or 10- shifts per week, as shown in Table C7.1-1.

Table C7.1-1 Estimated Operating Costs for RBOF and L-Basin (\$Millions)

Shifts Per Week	L-basin Annual Operations	RBOF Annual Operations
	Costs	Costs
5	9.5	8.2
10	12.0	10.5
15	14.6	12.8
21	17.2	15.0

If a treatment technology is not operational before 2004, an additional charge is assessed for installation of additional racks in L-basin. This is assumed to cost approximately \$5.0 million, comparable to the first increment of new racks at L-Basin.

7.2 Interim Dry Storage

Each treatment technology will require interim storage of its product between the time it is initially created, and the time the repository becomes available for shipments from SRS. Since the vast majority of the research reactor SNF will be delivered well before the repository is available for shipments, the interim storage facility will be sized based on the estimated quantities of canisters produced by the various treatment technologies.

Storage of Direct Disposal and Co-Disposal Canisters

The canister dimensions seem to make the modular dry vault interim storage concept very efficient, and this storage concept has been used as the baseline for cost estimating. This study

and previous studies have shown that all dry storage concepts are feasible and practical. The selection of the modular dry vault for a cost estimating baseline for this study in no way indicates a final selection of interim storage technology, or that other concepts may not be cost competitive or chosen during the facility design.

The cost basis for the modular vault is INEL reports EGG-WM-10670 Waste Management Facilities Cost Information for Spent Nuclear Fuel (March 1993), and EGG-WM-10708 Waste Management Facilities Cost Information Estimating Data for Spent Nuclear Fuel (March 1993). Module X from these references was used as the basis for capital costs, operating and maintenance costs, and decontamination and decommissioning costs.

Where possible based on the canister dimensions, it is assumed that more than one canister may be placed in a storage location. Two co-disposal canisters are assumed to be placed in each storage tube, and direct disposal packages are assumed to be stacked on top of each other subject to an overall height limit of 20 feet. The receipt and inspection function of module X was significantly scaled back since it is assumed that the interim storage facility will be constructed adjacent to the receipt and treatment facility. The storage vault portion of the facility costs were scaled based on the required storage capacity. The scaling relationship used is the exponential cost relationship described in DOE/SNF/REP-PS-001 Spent Nuclear Fuel Management Cost Evaluation Report (March 1995), Appendix C.

$$\$/\$_1 = (C_2/C_1)^R$$

where $\$_1$ and $\$_2$ are the costs for the base and scaled facilities, respectively, C_1 and C_2 are the capacities for the base and scaled facilities, and R is a scaling exponent, assumed to be 0.6 for this type of facility. Table C7.2-1 presents the required storage tubes for each option, the capacity ratio based on the Module X capacity of 230 storage tubes, and the associated cost scale factor. The Press and Dilute option considers requirements for dilution to either 20% enrichment or to 2% enrichment, resulting in either 400 or 1300 co-disposal canisters, respectively.

The canister storage facility estimate for the direct disposal phase of the reprocessing option is discussed in Section 8, since the storage facility is constructed integral with receiving and transfer functions.

Table C7.2-1 Required Interim Storage Tubes for Each Technology

Technology	Required Storage Tubes	Required Capacity Ratio	Cost Scale Factor
Direct Disposal	573	2.49	1.73
Co-Disposal	700	3.04	1.95
Press and Dilute (20%)	200	0.87	0.92
Press and Dilute (2%)	650	2.83	1.86
Melt and Dilute	200	0.87	0.92
Plasma Arc	200	0.87	0.92
GMODS	400	1.74	1.39
Dissolve and Vitrify	400	1.74	1.39
Electrometallurgical	0	N/A	N/A
Processing in Canyons	150	0.65	0.77

Tables C7.2-2,1 through C7.2-2,4 present details of the cost estimates for the interim storage facilities for the direct disposal option, including design, construction, operation, and decommissioning. These tables demonstrate the basis of the estimate, and the use of the cost scaling factors shown above. Tables C7.2-3 through C7.2-9 summarize total implementation, operation, and decommissioning costs for the other interim storage facilities.

Storage of Glass Logs

The Electrometallurgical and Processing options produce glass logs containing fission products but no fissile material. These packages are assumed to be stored at DWPF in one of the Glass Waste Storage Buildings (GWSB). The cost for storing these glass logs is estimated as an incremental increase of the cost of the construction of the next GWSB facility. Based on a current estimate of \$100 million total estimated cost, an additional 30% was added for Other Project Costs (OPC), for a total project cost (without contingency) of \$130 million for a facility for storage of approximately 2300 glass logs. Using the same scaling relationship described for the vault storage facilities, the incremental costs for adding 90 or 120 more storage locations was evaluated as \$3.0 million for the Electrometallurgical option, and \$4.0 million for the reprocessing option, respectively. This charge is assessed in 2004, when the next GWSB is expected to be constructed. Only capital costs are included, since incremental operating costs are negligible on this basis.

**Table C7.2-2 Storage Facility Estimate - Direct Disposal
Sheet 1**

Item	Description	Estimated Cost
DEVELOPMENT COSTS		
0.0	Development	
0.1	Research and Development	0
0.2	Process Demonstration	0
0.3	Waste Form Qualification	0
0.4	Development Contingency (0% of 0.1, 0.2, and 0.3)	0
	SUBTOTAL 0.0 DEVELOPMENT	0

IMPLEMENTATION COSTS		
1.0	Construction Costs (Fixed Price Contract)	
1.1	Building Structure Costs	9,212
1.2	Building Equipment Costs	12,766
1.3	Construction Indirects (29% of 1.1 and 1.2)	6,374
	SUBTOTAL Construction 1.0	28,352
2.0	Title I and Title II Design (8% of 1.0)	2,268
3.0	Inspection (3% of 1.0)	851
4.0	Project Support (See Note 1) (6% of 1.0)	1,701
5.0	Construction Management (14% of 1.0)	3,969
6.0	SUBTOTAL Engr & Const (1.0, 2.0, 3.0, 4.0, 5.0)	37,141
7.0	Site G & A (16% of 6.0)	5,943
8.0	Total Estimated Cost (TEC) (6.0, 7.0)	43,083
9.0	OPC (Other Project Costs) (See Note 2) (30% of 8.0)	12,925
10.0	Implementation Contingency (0% of 8.0 and 9.0)	
11.0	Total Projected Cost (8.0, 9.0, 10.0) IMPLEMENTATION	56,008

OPERATIONS COSTS					
12.0	Operations and Maintenance Costs (See Note 3)	5 Shifts / Week	10 Shifts / Week	15 Shifts / Week	21 Shifts / Week
12.1	Annual Operation Costs (Labor)	145	213	282	364
12.2	Annual Utility Costs	3	4	5	7
12.3	Annual Material / Consumable Costs	2	3	3	4
12.4	Annual Maintenance Costs	1,001	1,578	2,155	2,847
12.5	Interim Storage Costs	Included Above	Included Above	Included Above	Included Above
12.6	Operations Contingency (0% of 12.1 through 12.5)				
	SUBTOTAL 12.0 OPERATIONS (12.1 through 12.6).	1,151	1,798	2,445	3,222

DECONTAMINATION AND DECOMMISSIONING COSTS		
13.0	Decontamination and Decommissioning Costs	
13.1	Facility and Equipment D&D	
13.2	Storage D&D	2326
	SUBTOTAL 13.0 D&D	2326

NOTES:

1. Project Support includes Project Management, Project Control, Estimating and QA
2. Other Project Costs include NEPA process, developing operating procedures, startup and checkout.
3. Annual costs may vary with years depending on operations performed.
4. Percentages shown above are suggested starting point values, and may be adjusted.

**Table C7.2-2 Storage Facility Estimate - Direct Disposal
Sheet 2**

Building and Equipment Material and Installation Summary

	Area Description	Small Module Vault Facil Using EGG-WM-10708			Capacity Ratios		Scale Factor		Storage Facility for RRTF		
		Building	Equip	Total	Building	Equip	Building	Equip	Building	Equip	Total
X- 1	Receive and Inspect	258	150	408			0.2	0.2	52	30	82
X- 2	Wash Vehicle and Cask	374	541	915			0	0	0	0	0
X- 3	Cask Staging and Handling	186	2360	2546	1	1	1	1	186	2360	2546
X- 4	Canister Load and Unload	56	4823	4879	1	1	1	1	56	4823	4879
X- 5	Canister Transport	56		56	1	1	1	1	56	0	56
X- 6	Storage Cavity	4424	2149	6573	2.489	2.489	1.728	1.728	7646	3714	11360
X- 7	Monitor and Inspect	379	82	461	1	1	1	1	379	82	461
X- 8	Liquid Waste Treatment	31	475	506			0	0	0	0	0
X- 9	Decon & Survey	30	171	201			0.5	0.5	15	86	101
X- 10	Others		732	732	1	1	1	1	0	732	732
Total (1993 Dollars)		5794	11283	17077					8390	11627	20016

Note: Costs above are in 1993 Dollars. Convert to 1996 dollars multiplying by 1.098. ----->>>

9212	12766	21978
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93-94 94-95 95-96 Cum
102.60% 103.10% 103.80% 109.80%

Storage Capacity Required 573 tubes compared to 230 tubes in reference. Small Storage Module X (EGG-WM-10708)

- Notes:
1. Receipt & Inspection, Wash Vehicles and Cask, have been significantly reduced since this facility will be constructed adjacent to Transfer Facility.
 2. Cask delivery and staging areas not scaled back since there are only two delivery areas, and we want that configuration.
 3. Decon and survey reduced since with transfers inside a facility, surveys will be nominal, but general support rad protect will still be needed.

**Table C7.2-2 Storage Facility Estimate - Direct Disposal
Sheet 3**

Operating and Maintenance Summary (Loading, Unloading)

	Area Description	Small Storage Module X (EGG-WM-10708)				Scaling		Storage Facility O&M Est (15 shifts)				Storage Facility O&M Est (5 shifts)				Storage Facility O&M Est (10 shifts)				Storage Facility O&M Est (21 shifts)																																																	
		Maint	Utilities	Materials	FTE	Cap Ratio	Scale Factor	Maint	Utilities	Materials	FTE	Maint	Utilities	Materials	FTE	Maint	Utilities	Materials	FTE	Maint	Utilities	Materials	FTE																																														
X-1	Receive and Inspect	28	1	5	0.3		0.200	6	0	1	0.1	2	0	0	0.0	4	0	1	0.0	8	0	1	0.1																																														
X-2	Wash Vehicle and Cask	77	1				0.000	0	0	0	0.0	0	0	0	0.0	0	0	0	0.0	0	0	0	0.0																																														
X-3	Cask Staging and Handling	448	1		0.3	1.000	1.000	448	1	0	0.3	149	0	0	0.1	299	1	0	0.2	627	1	0	0.4																																														
X-4	Canister Load and Unload	644	1		0.3	1.000	1.000	644	1	0	0.3	215	0	0	0.1	429	1	0	0.2	902	1	0	0.4																																														
X-5	Canister Transport				0.3	1.000	1.000	0	0	0	0.3	0	0	0	0.1	0	0	0	0.2	0	0	0	0.4																																														
X-6	Storage Cavity	200	1		0.1	2.489	1.728	346	2	0	0.2	115	1	0	0.1	230	1	0	0.1	484	2	0	0.2																																														
X-7	Monitor and Inspect	18		1	0.2	1.000	1.000	18	0	1	0.2	6	0	0	0.1	12	0	1	0.1	25	0	1	0.3																																														
X-8	Liquid Waste Treatment	84	1	5			0.000	0	0	0	0.0	0	0	0	0.0	0	0	0	0.0	0	0	0	0.0																																														
X-9	Decon & Survey	25			0.0		0.500	13	0	0	0.0	4	0	0	0.0	8	0	0	0.0	18	0	0	0.0																																														
X-10	Others	102				1.000	1.000	102	0	0	0.0	34	0	0	0.0	68	0	0	0.0	143	0	0	0.0																																														
Total (1993 Dollars)		1626	6	11	210			1578	4	2	187	525	1	1	62	1051	3	1	124	2206	5	3	261																																														
		FTE (K\$) 140				Grand Total				1768 Grand Total				589 Grand Total				1179 Grand Total				2476																																															
Note: Costs above are in 1993 Dollars. Convert to 1996 dollars multiplying by 1.098. =====						1730				4				2				205				577				1				1				68				1153				3				1				137				2422				6				3				287			
						Grand Total (\$1996)				1942 Grand Total (\$1996)				647 Grand Total (\$1996)				1294 Grand Total (\$1996)				2718																																															

Operating and Maintenance Summary Loading (Storage)

	Area Description	Small Storage Module X (EGG-WM-10708)				Scaling		Storage Facility O&M Est (15 shifts)				Storage Facility O&M Est (5 shifts)				Storage Facility O&M Est (10 shifts)				Storage Facility O&M Est (21 shifts)																																	
		Maint	Utilities	Materials	FTE	Cap Ratio	Scale Factor	Maint	Utilities	Materials	FTE	Maint	Utilities	Materials	FTE	Maint	Utilities	Materials	FTE	Maint	Utilities	Materials	FTE																														
X-1	Receive and Inspect	6					0.200	1	0	0	0.0	1	0	0	0.0	1	0	0	0.0	1	0	0	0.0																														
X-2	Wash Vehicle and Cask	20					0.000	0	0	0	0.0	0	0	0	0.0	0	0	0	0.0	0	0	0	0.0																														
X-3	Cask Staging and Handling	110				1.000	1.000	110	0	0	0.0	110	0	0	0.0	110	0	0	0.0	110	0	0	0.0																														
X-4	Canister Load and Unload	158				1.000	1.000	158	0	0	0.0	158	0	0	0.0	158	0	0	0.0	158	0	0	0.0																														
X-5	Canister Transport					1.000	1.000	0	0	0	0.0	0	0	0	0.0	0	0	0	0.0	0	0	0	0.0																														
X-6	Storage Cavity	50				2.489	1.728	86	0	0	0.0	86	0	0	0.0	86	0	0	0.0	86	0	0	0.0																														
X-7	Monitor and Inspect	4	1	1	0.5	1.000	1.000	4	1	1	0.5	4	1	1	0.5	4	1	1	0.5	4	1	1	0.5																														
X-8	Liquid Waste Treatment	20					0.000	0	0	0	0.0	0	0	0	0.0	0	0	0	0.0	0	0	0	0.0																														
X-9	Decon & Survey	6					0.500	3	0	0	0.0	3	0	0	0.0	3	0	0	0.0	3	0	0	0.0																														
X-10	Others	24				1.000	1.000	24	0	0	0.0	24	0	0	0.0	24	0	0	0.0	24	0	0	0.0																														
Total (1993 Dollars)		398	1	1	70			387	1	1	70	387	1	1	70	387	1	1	70	387	1	1	70																														
		FTE (K\$) 140				Grand Total				459 Grand Total				459 Grand Total				459 Grand Total				459 Grand Total																															
Note: Costs above are in 1993 Dollars. Convert to 1996 dollars multiplying by 1.098. =====						425				1				1				77				425				1				1				77				425				1				1				77			
						Grand Total (\$1996)				504 Grand Total (\$1996)				504 Grand Total (\$1996)				504 Grand Total (\$1996)				504 Grand Total (\$1996)																															

Table C7.2-3 Storage Facility Estimate - Co - Disposal

Item	Description	Estimated Cost				
DEVELOPMENT COSTS						
0.0	Development					
0.1	Research and Development	0				
0.2	Process Demonstration	0				
0.3	Waste Form Qualification	0				
0.4	Development Contingency (0% of 0.1, 0.2, and 0.3)	0				
	SUBTOTAL 0.0 DEVELOPMENT	0				
IMPLEMENTATION COSTS						
1.0	Construction Costs (Fixed Price Contract)					
1.1	Building Structure Costs	10,288				
1.2	Building Equipment Costs	13,289				
1.3	Construction Indirects (29% of 1.1 and 1.2)	6,837				
	SUBTOTAL Construction 1.0	30,415				
2.0	Title I and Title II Design (8% of 1.0)	2,433				
3.0	Inspection (3% of 1.0)	912				
4.0	Project Support (See Note 1) (6% of 1.0)	1,825				
5.0	Construction Management (14% of 1.0)	4,258				
6.0	SUBTOTAL Engr & Const (1.0, 2.0, 3.0, 4.0, 5.0)	39,844				
7.0	Site G & A (16% of 6.0)	6,375				
8.0	Total Estimated Cost (TEC) (6.0, 7.0)	46,219				
9.0	OPC (Other Project Costs) (See Note 2) (30% of 8.0)	13,866				
10.0	Implementation Contingency (0% of 8.0 and 9.0)					
11.0	Total Projected Cost (8.0, 9.0, 10.0) IMPLEMENTATION	60,084				
OPERATIONS COSTS						
12.0	Operations and Maintenance Costs (See Note 3)		5 Shifts / Week	10 Shifts / Week	15 Shifts / Week	21 Shifts / Week
12.1	Annual Operation Costs (Labor)		146	216	285	368
12.2	Annual Utility Costs		3	4	6	7
12.3	Annual Material / Consumable Costs		2	3	3	4
12.4	Annual Maintenance Costs	1,030		1,623	2,216	2,927
12.5	Interim Storage Costs		Included Above	Included Above	Included Above	Included Above
12.6	Operations Contingency (0% of 12.1 through 12.5)					
	SUBTOTAL 12.0 OPERATIONS (12.1 through 12.6)	1,180	1,180	1,845	2,510	3,307
DECONTAMINATION AND DECOMMISSIONING COSTS						
13.0	Decontamination and Decommissioning Costs					
13.1	Facility and Equipment D&D					
13.2	Storage D&D	2571				
	SUBTOTAL 13.0 D&D	2571				

NOTES:

1. Project Support includes Project Management, Project Control, Estimating and QA
2. Other Project Costs include NEPA process, developing operating procedures, startup and checkout.
3. Annual costs may vary with years depending on operations performed.
4. Percentages shown above are suggested starting point values, and may be adjusted.

Table C7.2-4 Storage Facility Estimate - Press & Dilute 2%

Item	Description	Estimated Cost
------	-------------	----------------

DEVELOPMENT COSTS		
0.0	Development	
0.1	Research and Development	0
0.2	Process Demonstration	0
0.3	Waste Form Qualification	0
0.4	Development Contingency (0% of 0.1, 0.2, and 0.3)	0
	SUBTOTAL 0.0 DEVELOPMENT	0

IMPLEMENTATION COSTS		
1.0	Construction Costs (Fixed Price Contract)	
1.1	Building Structure Costs	9,877
1.2	Building Equipment Costs	13,089
1.3	Construction Indirects (29% of 1.1 and 1.2)	6,660
	SUBTOTAL Construction 1.0	29,625
2.0	Title I and Title II Design (8% of 1.0)	2,370
3.0	Inspection (3% of 1.0)	889
4.0	Project Support (See Note 1) (6% of 1.0)	1,778
5.0	Construction Management (14% of 1.0)	4,148
6.0	SUBTOTAL Engr & Const (1.0, 2.0, 3.0, 4.0, 5.0)	38,809
7.0	Site G & A (16% of 6.0)	6,209
8.0	Total Estimated Cost (TEC) (6.0, 7.0)	45,019
9.0	OPC (Other Project Costs) (See Note 2) (30% of 8.0)	13,506
10.0	Implementation Contingency (0% of 8.0 and 9.0)	
11.0	Total Projected Cost (8.0, 9.0, 10.0) IMPLEMENTATION	58,524

OPERATIONS COSTS					
		5 Shifts / Week	10 Shifts / Week	15 Shifts / Week	21 Shifts / Week
12.0	Operations and Maintenance Costs (See Note 3)				
12.1	Annual Operation Costs (Labor)	146	215	284	367
12.2	Annual Utility Costs	3	4	6	7
12.3	Annual Material / Consumable Costs	2	3	3	4
12.4	Annual Maintenance Costs	1,019	1,606	2,192	2,896
12.5	Interim Storage Costs	Included Above	Included Above	Included Above	Included Above
12.6	Operations Contingency (0% of 12.1 through 12.5)				
	SUBTOTAL 12.0 OPERATIONS (12.1 through 12.6).	1,169	1,827	2,485	3,275

DECONTAMINATION AND DECOMMISSIONING COSTS		
13.0	Decontamination and Decommissioning Costs	
13.1	Facility and Equipment D&D	
13.2	Storage D&D	2477
	SUBTOTAL 13.0 D&D	2477

NOTES:

1. Project Support includes Project Management, Project Control, Estimating and QA
2. Other Project Costs include NEPA process, developing operating procedures, startup and checkout.
3. Annual costs may vary with years depending on operations performed.
4. Percentages shown above are suggested starting point values, and may be adjusted.

Table C7.2-5 Storage Facility Estimate - Press & Dilute 20%

Item	Description	Estimated Cost
------	-------------	----------------

DEVELOPMENT COSTS		
0.0	Development	
0.1	Research and Development	0
0.2	Process Demonstration	0
0.3	Waste Form Qualification	0
0.4	Development Contingency (0% of 0.1, 0.2, and 0.3)	
	SUBTOTAL 0.0 DEVELOPMENT	0

IMPLEMENTATION COSTS		
1.0	Construction Costs (Fixed Price Contract)	
1.1	Building Structure Costs	5,283
1.2	Building Equipment Costs	10,858
1.3	Construction Indirects (29% of 1.1 and 1.2)	4,681
	SUBTOTAL Construction 1.0	20,822
2.0	Title I and Title II Design (8% of 1.0)	1,666
3.0	Inspection (3% of 1.0)	625
4.0	Project Support (See Note 1) (6% of 1.0)	1,249
5.0	Construction Management (14% of 1.0)	2,915
	SUBTOTAL Engr & Const (1.0, 2.0, 3.0, 4.0, 5.0)	27,277
7.0	Site G & A (16% of 6.0)	4,364
8.0	Total Estimated Cost (TEC) (6.0, 7.0)	31,641
9.0	OPC (Other Project Costs) (See Note 2) (30% of 8.0)	9,492
10.0	Implementation Contingency (0% of 8.0 and 9.0)	
11.0	Total Projected Cost (8.0, 9.0, 10.0) IMPLEMENTATION	41,133

OPERATIONS COSTS					
12.0	Operations and Maintenance Costs (See Note 3)	5 Shifts / Week	10 Shifts / Week	15 Shifts / Week	21 Shifts / Week
12.1	Annual Operation Costs (Labor)	141	205	269	346
12.2	Annual Utility Costs	2	3	5	6
12.3	Annual Material / Consumable Costs	2	3	3	4
12.4	Annual Maintenance Costs	898	1,415	1,933	2,554
12.5	Interim Storage Costs	Included Above	Included Above	Included Above	Included Above
12.6	Operations Contingency (0% of 12.1 through 12.5)				
	SUBTOTAL 12.0 OPERATIONS (12.1 through 12.6)	1,043	1,626	2,210	2,910

DECONTAMINATION AND DECOMMISSIONING COSTS		
13.0	Decontamination and Decommissioning Costs	
13.1	Facility and Equipment D&D	
13.2	Storage D&D	1430
	SUBTOTAL 13.0 D&D	1430

NOTES:

1. Project Support includes Project Management, Project Control, Estimating and QA
2. Other Project Costs include NEPA process, developing operating procedures, startup and checkout.
3. Annual costs may vary with years depending on operations performed.
4. Percentages shown above are suggested starting point values, and may be adjusted.

Table C7.2-6 Storage Facility Estimate - Melt & Dilute

Item	Description	Estimated Cost
DEVELOPMENT COSTS		
0.0	Development	
0.1	Research and Development	0
0.2	Process Demonstration	0
0.3	Waste Form Qualification	0
0.4	Development Contingency (0% of 0.1, 0.2, and 0.3)	0
	SUBTOTAL 0.0 DEVELOPMENT	0

IMPLEMENTATION COSTS		
1.0	Construction Costs (Fixed Price Contract)	
1.1	Building Structure Costs	5,283
1.2	Building Equipment Costs	10,858
1.3	Construction Indirects (29% of 1.1 and 1.2)	4,681
	SUBTOTAL Construction 1.0	20,822
2.0	Title I and Title II Design (8% of 1.0)	1,666
3.0	Inspection (3% of 1.0)	625
4.0	Project Support (See Note 1) (6% of 1.0)	1,249
5.0	Construction Management (14% of 1.0)	2,915
6.0	SUBTOTAL Engr & Const (1.0, 2.0, 3.0, 4.0, 5.0)	27,277
7.0	Site G & A (16% of 6.0)	4,364
8.0	Total Estimated Cost (TEC) (6.0, 7.0)	31,641
9.0	OPC (Other Project Costs) (See Note 2) (30% of 8.0)	9,492
10.0	Implementation Contingency (0% of 8.0 and 9.0)	
11.0	Total Projected Cost (8.0, 9.0, 10.0) IMPLEMENTATION	41,133

OPERATIONS COSTS					
		5 Shifts / Week	10 Shifts / Week	15 Shifts / Week	21 Shifts / Week
12.0	Operations and Maintenance Costs (See Note 3)				
12.1	Annual Operation Costs (Labor)	141	205	269	346
12.2	Annual Utility Costs	2	3	5	6
12.3	Annual Material / Consumable Costs	2	3	3	4
12.4	Annual Maintenance Costs	898	1,415	1,933	2,554
12.5	Interim Storage Costs	Included Above	Included Above	Included Above	Included Above
12.6	Operations Contingency (0% of 12.1 through 12.5)				
	SUBTOTAL 12.0 OPERATIONS (12.1 through 12.6).	1,043	1,626	2,210	2,910

DECONTAMINATION AND DECOMMISSIONING COSTS		
13.0	Decontamination and Decommissioning Costs	
13.1	Facility and Equipment D&D	
13.2	Storage D&D	1430
	SUBTOTAL 13.0 D&D	1430

NOTES:

1. Project Support includes Project Management, Project Control, Estimating and QA
2. Other Project Costs include NEPA process, developing operating procedures, startup and checkout.
3. Annual costs may vary with years depending on operations performed.
4. Percentages shown above are suggested starting point values, and may be adjusted.

Table C7.2-7 Storage Facility Estimate - Plasma Arc

Item	Description	Estimated Cost
DEVELOPMENT COSTS		
0.0	Development	
0.1	Research and Development	0
0.2	Process Demonstration	0
0.3	Waste Form Qualification	0
0.4	Development Contingency (0% of 0.1, 0.2, and 0.3)	0
	SUBTOTAL 0.0 DEVELOPMENT	0
IMPLEMENTATION COSTS		
1.0	Construction Costs (Fixed Price Contract)	
1.1	Building Structure Costs	5,283
1.2	Building Equipment Costs	10,858
1.3	Construction Indirects (29% of 1.1 and 1.2)	4,681
	SUBTOTAL Construction 1.0	20,822
2.0	Title I and Title II Design (8% of 1.0)	1,666
3.0	Inspection (3% of 1.0)	625
4.0	Project Support (See Note 1) (6% of 1.0)	1,249
5.0	Construction Management (14% of 1.0)	2,915
6.0	SUBTOTAL Engr & Const (1.0, 2.0, 3.0, 4.0, 5.0)	27,277
7.0	Site G & A (16% of 6.0)	4,364
8.0	Total Estimated Cost (TEC) (6.0, 7.0)	31,641
9.0	OPC (Other Project Costs) (See Note 2) (30% of 8.0)	9,492
10.0	Implementation Contingency (0% of 8.0 and 9.0)	
11.0	Total Projected Cost (8.0, 9.0, 10.0) IMPLEMENTATION	41,133
OPERATIONS COSTS		
12.0	Operations and Maintenance Costs (See Note 3)	
12.1	Annual Operation Costs (Labor)	141
12.2	Annual Utility Costs	205
12.3	Annual Material / Consumable Costs	3
12.4	Annual Maintenance Costs	2
12.5	Interim Storage Costs	898
12.6	Operations Contingency (0% of 12.1 through 12.5)	Included Above
	SUBTOTAL 12.0 OPERATIONS (12.1 through 12.6).	1,043
		5 Shifts / Week
		10 Shifts / Week
		15 Shifts / Week
		21 Shifts / Week
		346
		6
		4
		2,554
		Included Above
		2,910
DECONTAMINATION AND DECOMMISSIONING COSTS		
13.0	Decontamination and Decommissioning Costs	
13.1	Facility and Equipment D&D	
13.2	Storage D&D	1430
	SUBTOTAL 13.0 D&D	1430

NOTES:

1. Project Support includes Project Management, Project Control, Estimating and QA
2. Other Project Costs include NEPA process, developing operating procedures, startup and checkout.
3. Annual costs may vary with years depending on operations performed.
4. Percentages shown above are suggested starting point values, and may be adjusted.

Table C7.2-8 Storage Facility Estimate - GMODs

Item	Description	Estimated Cost				
DEVELOPMENT COSTS						
0.0	Development					
0.1	Research and Development	0				
0.2	Process Demonstration	0				
0.3	Waste Form Qualification	0				
0.4	Development Contingency (0% of 0.1, 0.2, and 0.3)	0				
	SUBTOTAL 0.0 DEVELOPMENT	0				
IMPLEMENTATION COSTS						
1.0	Construction Costs (Fixed Price Contract)					
1.1	Building Structure Costs	7,587				
1.2	Building Equipment Costs	11,977				
1.3	Construction Indirects (29% of 1.1 and 1.2)	5,673				
	SUBTOTAL Construction 1.0	25,237				
2.0	Title I and Title II Design (8% of 1.0)	2,019				
3.0	Inspection (3% of 1.0)	757				
4.0	Project Support (See Note 1) (6% of 1.0)	1,514				
5.0	Construction Management (14% of 1.0)	3,533				
6.0	SUBTOTAL Engr & Const (1.0, 2.0, 3.0, 4.0, 5.0)	33,061				
7.0	Site G & A (16% of 6.0)	5,290				
8.0	Total Estimated Cost (TEC) (6.0, 7.0)	38,350				
9.0	OPC (Other Project Costs) (See Note 2) (30% of 8.0)	11,505				
10.0	Implementation Contingency (0% of 8.0 and 9.0)					
11.0	Total Projected Cost (8.0, 9.0, 10.0) IMPLEMENTATION	49,855				
OPERATIONS COSTS						
12.0	Operations and Maintenance Costs (See Note 3)		5 Shifts / Week	10 Shifts / Week	15 Shifts / Week	21 Shifts / Week
12.1	Annual Operation Costs (Labor)	143	210	277	356	
12.2	Annual Utility Costs	2	4	5	7	
12.3	Annual Material / Consumable Costs	2	3	3	4	
12.4	Annual Maintenance Costs	958	1,511	2,063	2,726	
12.5	Interim Storage Costs	Included Above	Included Above	Included Above	Included Above	
12.6	Operations Contingency (0% of 12.1 through 12.5)					
	SUBTOTAL 12.0 OPERATIONS (12.1 through 12.6).	1,106	1,727	2,348	3,093	
DECONTAMINATION AND DECOMMISSIONING COSTS						
13.0	Decontamination and Decommissioning Costs					
13.1	Facility and Equipment D&D					
13.2	Storage D&D	1955				
	SUBTOTAL 13.0 D&D	1955				

NOTES:

1. Project Support includes Project Management, Project Control, Estimating and QA
2. Other Project Costs include NEPA process, developing operating procedures, startup and checkout.
3. Annual costs may vary with years depending on operations performed.
4. Percentages shown above are suggested starting point values, and may be adjusted.

Table C7.2-9 Storage Facility Estimate - D&V

Item	Description	Estimated Cost			
DEVELOPMENT COSTS					
0.0	Development				
0.1	Research and Development	0			
0.2	Process Demonstration	0			
0.3	Waste Form Qualification	0			
0.4	Development Contingency (0% of 0.1, 0.2, and 0.3)	0			
	SUBTOTAL 0.0 DEVELOPMENT	0			
IMPLEMENTATION COSTS					
1.0	Construction Costs (Fixed Price Contract)				
1.1	Building Structure Costs	7,587			
1.2	Building Equipment Costs	11,977			
1.3	Construction Indirects (29% of 1.1 and 1.2)	5,673			
	SUBTOTAL Construction 1.0	25,237			
2.0	Title I and Title II Design (8% of 1.0)	2,019			
3.0	Inspection (3% of 1.0)	757			
4.0	Project Support (See Note 1) (6% of 1.0)	1,514			
5.0	Construction Management (14% of 1.0)	3,533			
6.0	SUBTOTAL Engr & Const (1.0, 2.0, 3.0, 4.0, 5.0)	33,061			
7.0	Site G & A (16% of 6.0)	5,290			
8.0	Total Estimated Cost (TEC) (6.0, 7.0)	38,350			
9.0	OPC (Other Project Costs) (See Note 2) (30% of 8.0)	11,505			
10.0	Implementation Contingency (0% of 8.0 and 9.0)				
11.0	Total Projected Cost (8.0, 9.0, 10.0) IMPLEMENTATION	49,855			
OPERATIONS COSTS					
12.0	Operations and Maintenance Costs (See Note 3)				
12.1	Annual Operation Costs (Labor)				
12.2	Annual Utility Costs	143	210	277	356
12.3	Annual Material / Consumable Costs	2	4	5	7
12.4	Annual Maintenance Costs	2	3	3	4
12.5	Interim Storage Costs	958	1,511	2,063	2,726
12.6	Operations Contingency (0% of 12.1 through 12.5)	Included Above	Included Above	Included Above	Included Above
	SUBTOTAL 12.0 OPERATIONS (12.1 through 12.6).	1,106	1,727	2,348	3,093
DECONTAMINATION AND DECOMMISSIONING COSTS					
13.0	Decontamination and Decommissioning Costs				
13.1	Facility and Equipment D&D				
13.2	Storage D&D	1955			
	SUBTOTAL 13.0 D&D	1955			

NOTES:

1. Project Support includes Project Management, Project Control, Estimating and QA
2. Other Project Costs include NEPA process, developing operating procedures, startup and checkout.
3. Annual costs may vary with years depending on operations performed.
4. Percentages shown above are suggested starting point values, and may be adjusted.

7.3 Disposal

For the purposes of this study, disposal charges are composed of transportation to the repository, and repository fees and operations costs.

Transport to Repository

Using the DOE Office of Civilian Radioactive Waste Management report DOE/RW-0479, Analysis of the Total System Life Cycle Cost of the Civilian Radioactive Waste Management Program as a reference, the cost of transportation from SRS to the repository can be estimated as \$160,000 per shipment. This is an "all-in" cost, including development of cask systems, operations, and decommissioning at the end of useful life. This figure is not sensitive to treatment technology, and it is assumed that cask configurations similar to defense HLW shipping casks will be developed. Four DHLW canisters are assumed to be placed in each cask. For the small (24" x 5') direct disposal canister, it is assumed that 12 canisters can be placed in one shipping cask. For the co-disposal canisters, it is assumed that a specialty cask will be designed and procured so that seven canisters can be carried in each shipment. Applying these unit costs to the quantity of canisters estimated for the various technologies results in the total transportation costs shown in Table 7.3-1. These total costs are distributed in time so that 80% of the shipments occur in the first eight years after repository shipments begin, and the remaining 20% are uniformly distributed over the remaining eight years before end of research reactor SNF operations at SRS.

Table 7.3-1 Transportation Operations Costs (Millions)

Technology	Number of Shipments	Transportation Operations Cost
Direct Disposal	167	\$26.7
Co-Disposal	200	\$32.0
Press and Dilute (20%)	57	\$9.1
Press and Dilute (2%)	186	\$29.7
Melt and Dilute	57	\$9.1
Plasma Arc	57	\$9.1
GMODS	114	\$18.3
Dissolve and Vitrify	114	\$18.3
Electrometallurgical	23	\$3.6
Processing in Canyons	74	\$11.8

Repository Fee and Repository Operations.

Costs associated with the disposal of research reactor SNF can be divided into those that are relatively fixed, and do not change with the number of waste packages, and those costs which are directly proportional to the number of waste packages emplaced. Costs in the first category include the development and licensing of the repository and associated support systems. For the purpose of this study, this fixed repository development fee has been estimated based on the ratio of the volume of research reactor SNF to the volume of all SNF in the repository, times the total projected cost of the repository. This figure is \$30 million for all technologies evaluated, and represents an estimate of DOE's share of repository costs, based on the volume of DOE SNF to be stored. Since the same figure is used for all technologies, the repository fee is not a discriminator in the selection of a disposal technology.

Other direct costs, disposal operations costs including the cost of the individual waste package itself, handling of the individual waste packages, transportation underground and emplacement in the repository drift, are directly proportional to the number of waste packages. Three concepts for disposal are included in the current study: direct disposal of small packages; co-disposal inside defense HLW packages, and disposal of DWPf canisters as defense HLW. Estimated unit costs per repository waste package are shown in Table C7.3-2.

Table C7.3-2 Repository Waste Package Unit Costs

Package Type	Package Cost	Surface and Subsurface Operations Cost	Total Cost
24" x 10' DD Can	\$180,000	\$170,000	\$350,000
24" x 16' DD Can	\$300,000	\$180,000	\$480,000
24" x 5' DD Can	\$100,000	\$150,000	\$250,000
Defense HLW 5-Pack	\$400,000	\$200,000	\$600,000 ¹

¹ For the co-disposal concepts, the repository costs for a given defense HLW waste package are divided equally between the five HLW glass canisters and the co-disposed 17" x 10' research reactor SNF canister. That is to say, that \$100,000 is apportioned to each of the five glass logs in the HLW waste package, and the last \$100,000 is apportioned to the co-disposed research reactor canister. For disposal of DHLW canisters, the HLW waste package cost is allocated to the five HLW canister slots in the waste package. That is, if there is no co-disposal container, each HLW canister is apportioned \$120,000 for the cost of the waste package.

By multiplying the quantities of waste packages in Table C3.0-4 by the unit costs in Table C7.3-2, the estimated repository operations costs in Table C7.3-3 result. These costs are distributed over the period of shipments from SRS to the repository.

Table C7.3-3 Repository Operations Costs (Millions)

Technology	Repository Operations Cost
Direct Disposal	\$368
Co-Disposal	\$140
Press and Dilute (20%)	\$40
Press and Dilute (2%)	\$130
Melt and Dilute	\$40
Plasma Arc	\$40
GMODS	\$80
Dissolve and Vitrify	\$80
Electrometallurgical	\$11
Processing in Canyons	\$45

7.4 Cost of Canisters

The cost of direct disposal and co-disposal canisters (packages) were estimated based on the OCRWM MPC unit costs (MPC CDR, Volume IV). The costs for DHLW canisters was estimated based on SRS experience with DWPF. The associated costs are shown in Table C7.4-1.

Table C7.4-1 Assumed Canister Costs

Canister Size	Cost (thousands of dollars)
24" x 10' Direct Disposal	\$44.50
24" x 16' Direct Disposal	\$66.75
24" x 5" Direct Disposal	\$22.25
17" x 10' Co-Disposal	\$21.00
24" x 10' DHLW Glass Log	\$10.00

The total costs for these packages is distributed throughout the years of operation based on the quantity of material treated each year.

7.5 Cost Adjustments

Hot Vacuum Drying Facility

In addition to the aluminum-based SNF which is the subject of this study, there is an existing inventory of uranium metal SNF at SRS. Some of the research reactor SNF treatment technologies evaluated can directly process uranium metal SNF. Other technologies, which do not change the chemical composition of the SNF, cannot resolve the dry storage issues associated with this material without additional treatment. For the technologies which do not alter the chemical composition of the SNF, a hot vacuum drying facility similar to that planned for Hanford is included in the cost estimate. The basis of the cost estimate is a scaling of the conceptual cost estimate prepared for Hanford for a green field version of the HVD facility. The scale factor is 44% based on a capacity ratio of 1/4, and the exponential scaling relationship described in Section 7.2. The resulting building and equipment cost is \$9.808 million, and the estimated annual operations cost is \$10 million resulting in a total cost of \$61 million for this additional treatment. Table C7.5-1 presents a summary of the costs associated with the HVD Facility, including associated site overheads. These costs are assumed to be deferred until after the peak construction expenditure years associated with the transfer facility and the treatment module.

It should be noted that WSRC-RP-95-798 estimates that the cost of reprocessing all uranium metal SNF would be approximately \$30 million. If this proves to be a viable alternative, construction of the HVD facility could be avoided, saving approximately \$30 million in total system costs.

Uranium Credit

Two of the treatment technologies considered separate out U-235 from other materials. This material is assumed to have a market value for commercial fuel fabricators, and it is assumed that the U-235 removed from the research reactor SNF would be sold at market rates. For the Processing option, this would result in a credit of \$175 million. The Electrometallurgical process will result in a purer form of the U-235 material, and the comparable credit will be \$223 million. These credits are distributed in time based on the amount of material treated each year.

Table C7.5-1 Summary of Hot Vacuum Drying Facility Costs

Item	Description	Estimated Cost
DEVELOPMENT COSTS		
0.0	Development	
0.1	Research and Development	0
0.2	Process Demonstration	0
0.3	Waste Form Qualification	0
0.4	Development Contingency (0% of 0.1, 0.2, and 0.3)	0
	SUBTOTAL 0.0 DEVELOPMENT	0
IMPLEMENTATION COSTS		
1.0	Construction Costs (Fixed Price Contract)	
1.1	Building Structure Costs	Included Below
1.2	Building Equipment Costs	9,808
1.3	Construction Indirects (29% of 1.1 and 1.2)	2,844
	SUBTOTAL Construction 1.0	12,652
2.0	Title I and Title II Design (8% of 1.0)	1,012
3.0	Inspection (3% of 1.0)	380
4.0	Project Support (See Note 1) (6% of 1.0)	759
5.0	Construction Management (14% of 1.0)	1,771
6.0	SUBTOTAL Engr & Const (1.0, 2.0, 3.0, 4.0, 5.0)	16,575
7.0	Site G & A (16% of 6.0)	2,652
8.0	Total Estimated Cost (TEC) (6.0, 7.0)	19,226
9.0	OPC (Other Project Costs) (See Note 2) (30% of 8.0)	5,768
10.0	Implementation Contingency (0% of 8.0 and 9.0)	
11.0	Total Projected Cost (8.0, 9.0, 10.0) IMPLEMENTATION	24,994
OPERATIONS COSTS		
12.0	Operations and Maintenance Costs (See Note 3)	10 Shifts / Week
12.1	Annual Operation Costs (Labor)	
12.2	Annual Utility Costs	
12.3	Annual Material / Consumable Costs	
12.4	Annual Maintenance Costs	
12.5	Interim Storage Costs	
12.6	Operations Contingency (0% of 12.1 through 12.5)	0
	SUBTOTAL 12.0 OPERATIONS (12.1 through 12.6).	10,000
	(Assumption Based On RBOF Operations)	
DECONTAMINATION AND DECOMMISSIONING COSTS		
13.0	Decontamination and Decommissioning Costs	
13.1	Facility and Equipment D&D	5,000
13.2	Storage D&D	
	SUBTOTAL 13.0 D&D	5,000

NOTES:

1. Project Support includes Project Management, Project Control, Estimating and QA
2. Other Project Costs include NEPA process, developing operating procedures, startup and checkout.
3. Annual costs may vary with years depending on operations performed.
4. Percentages shown above are suggested starting point values, and may be adjusted.

8.0 Transfer, Treatment and Packaging Costs

This section describes details of cost evaluations that are specific to the technology involved. Each treatment technology will involve the functions of cask receipt, SNF unloading, and loading and sealing products into packages. The structures and equipment for these functions will be very similar for any of the options. For the purposes of these cost evaluations, it is assumed that the same transfer and packaging facilities and equipment are provided for each option based on the direct disposal transfer facility cost model. A treatment module was conceptualized for each technology, which would be incorporated into the transfer facility structure during the detailed design process. The concept included a list of significant equipment associated with the treatment technology, and incremental building space needed for the treatment process. The building space and equipment costs were estimated, and these were added to the basic transfer facility costs for a total new facility cost.

8.1 Direct Disposal

General Description

Appendix D provides an overview of the direct disposal option, while Section 11.0 describes operating variations considered in conceptualizing the system. The central feature of this technology is the construction of a new dry transfer facility that would serve to receive research reactor SNF shipments from off-site or from existing basin storage, to transfer the SNF into packages designed for storage, transportation and disposal in the repository, to dry and seal these packages, and to place them into interim dry storage at SRS. These packages are designed for disposal as individual waste packages. When the repository becomes available for shipments of this type of material, the canisters are withdrawn from storage, loaded into transportation overpacks, and shipped to the repository. Based on commitments from DOE executive management, it is assumed that the funding and procurement process will be expedited so that adequate funding will be made available in 1998, so that this facility can be brought on-line in 2001.

Cost Estimate for Dry Transfer Facility

R&D and Concept Development

The technology for all aspects of this concept is already in place, and little research and development is needed. This concept can proceed directly to the design and construction of required facilities and equipment with a nominal R&D and development expenditure of \$10 million.

Facilities, Equipment and Operations

The cost basis for the dry transfer facility is INEL reports EGG-WM-10670 Waste Management Facilities Cost Information for Spent Nuclear Fuel (March 1993), and EGG-WM-10708 Waste Management Facilities Cost Information Estimating Data for Spent Nuclear Fuel (March 1993). Because of the similarity of functions performed and production capacity required, Module U

from these references was used as the basis for capital costs, operating and maintenance costs, and decontamination and decommissioning costs of the dry transfer facility. Systems which were determined to be in excess of that required for operation were scaled back. Table C8.1-1sh1 presents a summary of the total project costs for the transfer facility, including design, construction, operation and decommissioning, including appropriate funding overheads. Table C8.1sh2 through sh4 show the details of these estimates, and demonstrates the scaling of costs from the reference. These tables also show a listing of systems included in the facility, and the functions performed.

As a confirmatory point of reference, the OCRWM MRS transfer facility estimate was reviewed and scaled to the appropriate capacity for use with SRS SNF. This exercise produced the building and equipment cost estimate shown in Table C8.1-2. The two facility estimates are very close, and this confirms the reasonableness of the estimate based on the INEL reports.

Overall Summary of Costs

Table C8.1-3 shows the total system costs, and Table C8.1-4 present total system costs and their distribution in time.

**Table C8.1-1 Transfer Facility Estimate - Direct Disposal
Sheet 1**

Item	Description	Estimated Cost
------	-------------	----------------

DEVELOPMENT COSTS		
0.0	Development	
0.1	Research and Development	5,000
0.2	Process Demonstration	5,000
0.3	Waste Form Qualification	0
0.4	Development Contingency (0% of 0.1, 0.2, and 0.3)	
	SUBTOTAL 0.0 DEVELOPMENT	10,000

IMPLEMENTATION COSTS		
1.0	Construction Costs (Fixed Price Contract)	
1.1	Building Structure Costs	36,399
1.2	Building Equipment Costs	30,138
1.3	Construction Indirects (29% of 1.1 and 1.2)	19,296
	SUBTOTAL Construction 1.0	85,832
2.0	Title I and Title II Design (8% of 1.0)	6,867
3.0	Inspection (3% of 1.0)	2,575
4.0	Project Support (See Note 1) (6% of 1.0)	5,150
5.0	Construction Management (14% of 1.0)	12,017
6.0	SUBTOTAL Engr & Const (1.0, 2.0, 3.0, 4.0, 5.0)	112,440
7.0	Site G & A (16% of 6.0)	17,990
8.0	Total Estimated Cost (TEC) (6.0, 7.0)	130,431
9.0	OPC (Other Project Costs) (See Note 2) (30% of 8.0)	39,129
10.0	Implementation Contingency (0% of 8.0 and 9.0)	
11.0	Total Projected Cost (8.0, 9.0, 10.0) IMPLEMENTATION	169,560

OPERATIONS COSTS					
12.0	Operations and Maintenance Costs (See Note 3)	5 Shifts / Week	10 Shifts / Week	15 Shifts / Week	21 Shifts / Week
12.1	Annual Operation Costs (Labor)	460	920	1380	1932
12.2	Annual Utility Costs	72	144	216	303
12.3	Annual Material / Consumable Costs	374	748	1122	1570
12.4	Annual Maintenance Costs	2,151	4302	6454	9035
12.5	Interim Storage Costs				
12.6	Operations Contingency (0% of 12.1 through 12.5)				
	SUBTOTAL 12.0 OPERATIONS (12.1 through 12.6).	3,057	6,114	9,172	12,840

DECONTAMINATION AND DECOMMISSIONING COSTS		
13.0	Decontamination and Decommissioning Costs	
13.1	Facility and Equipment D&D	18,530
13.2	Storage D&D	
	SUBTOTAL 13.0 D&D	18,530

NOTES:

1. Project Support includes Project Management, Project Control, Estimating and QA
2. Other Project Costs include NEPA process, developing operating procedures, startup and checkout.
3. Annual costs may vary with years depending on operations performed.
4. Percentages shown above are suggested starting point values, and may be adjusted.

**Table C8.1-1 Transfer Facility Estimate - Direct Disposal
Sheet 2**

Building and Equipment Material and Installation Summary

	Area Description	Small C&S Facil Using EGG-WM-10708			Reduction WSRC-RP-95-798			Result WSRC-RP-95-798			Alternate Reduction for RRTF			Alternate Result for RRTF		
		Building	Equip	Total	Building	Equip	Total	Building	Equip	Total	Building	Equip	Total	Building	Equip	Total
U- 1	Receive and Inspect	389	386	775			0	389	386	775			0	389	386	775
U- 2	Wash Vehicle and Cask	3780	446	4226			0	3780	446	4226			0	3780	446	4226
U- 3	Cask Staging and Handling	3320	3163	6483			0	3320	3163	6483			0	3320	3163	6483
U- 4	SNF Loading	1328	3417	4745			0	1328	3417	4745			0	1328	3417	4745
U- 5	De-Canning	1328	2766	4094	400	1000	1400	928	1766	2694	400	1000	1400	928	1766	2694
U- 6	Sampling	1845	2691	4536	800	2691	3491	1045	0	1045	1845	2691	4536	0	0	0
U- 7	SNF Analysis	2835	1802	4637	400	900	1300	2435	902	3337	1400	900	2300	1435	902	2337
U- 8	Waste Treatment and Pkg	1659	3988	5647	200	1000	1200	1459	2988	4447	600	1000	1600	1059	2988	4047
U- 9	Conditioning	3431	3640	7071	200	1000	1200	3231	2640	5871	900	1000	1800	2631	2640	5271
U- 10	Canning	3873	1909	5782			0	3873	1909	5782			0	3873	1909	5782
U- 11	Canister and Decon Survey	3320	1847	5167			0	3320	1847	5167			0	3320	1847	5167
U- 12	Canister Loading	1660		1660			0	1660	0	1660			0	1660	0	1660
U- 13	Hardware and Scrap Metal Pkg.	218	567	783			0	218	567	783			0	218	567	783
U- 14	Util & Service Air / Water	216	584	800			0	216	584	800			0	216	584	800
U- 15	Emergency Power	1890	375	2265			0	1890	375	2265			0	1890	375	2265
U- 16	Heating Ventilating	162	902	1064			0	162	902	1064			0	162	902	1064
U- 17	Radiation Monitoring	216	1018	1234			0	216	1018	1234			0	216	1018	1234
U- 18	Electrical & Controls	5644	867	6511			0	5644	867	6511			0	5644	867	6511
U- 19	Maintenance Hot Cell	840	2305	3145			0	840	2305	3145			0	840	2305	3145
U- 20	Manipulator Maintenance	243	919	1162			0	243	919	1162			0	243	919	1162
U- 21	Fresh Container Del & Stor		402	402			0	0	402	402			0	0	402	402
U- 22	Others		45	45			0	0	45	45			0	0	45	45
Total (1993 Dollars)		38195	34039	72234	2000	6591	8591	38195	27448	65643	5045	6591	11636	33150	27448	60598

Note: Costs above are in 1993 Dollars. Convert to 1996 dollars multiplying by 1.098. ----->>>
 93-94 94-95 95-96 Cum
 102.60% 103.10% 103.80% 109.80%

69880

36399 30138 66537

**Table C8.1-1 Transfer Facility Estimate - Direct Disposal
Sheet 3**

Annual Operating and Maintenance Costs

	Area Description	Small C&S Facil Using EGG-WM-10708				Scaling		Transfer Facility O&M Est (15 shifts)				Transfer Facility O&M Est (21 shifts)				Transfer Facility O&M Est (5 shifts)				Transfer Facility O&M Est (10 shifts)			
		Maint	Utilities	Materials	FTE	Cap Ratio	Scale Factor	Maint	Utilities	Materials	FTE	Maint	Utilities	Materials	FTE	Maint	Utilities	Materials	FTE	Maint	Utilities	Materials	FTE
U- 1	Receive and Inspect	70	2	5	1.0	1.000	70	2	5	1.0	98	3	7	1.4	23	1	2	0	47	1	3	0.7	
U- 2	Wash Vehicle and Cask	84	12	2	0.5	1.000	84	12	2	0.5	118	17	3	0.7	28	4	1	0	56	8	1	0.3	
U- 3	Cask Staging and Handling	707	15	2	0.5	1.000	707	15	2	0.5	990	21	3	0.7	236	5	1	0	471	10	1	0.3	
U- 4	SNF Loading	760	7	2	0.5	1.000	760	7	2	0.5	1064	10	3	0.7	253	2	1	0	507	5	1	0.3	
U- 5	De-Canning	620	6		0.5	0.500	409	4	0	0.3	573	6	0	0.5	136	1	0	0	273	3	0	0.2	
U- 6	Sampling	574	4		0.5	0.000	0	0	0	0.0	0	0	0	0.0	0	0	0	0	0	0	0	0.0	
U- 7	SNF Analysis	378	2	30	0.5	0.300	0.498	184	1	15	0.2	257	1	20	0.3	61	0	5	0	122	1	10	0.2
U- 8	Waste Treatment and Pkg	896	15	150	0.5	0.600	0.736	652	11	110	0.4	913	15	155	0.5	217	4	37	0	435	7	74	0.2
U- 9	Conditioning	788	8	5	1.0	0.600	0.736	590	6	4	0.7	812	8	5	1.0	193	2	1	0	387	4	2	0.5
U- 10	Canning	403	8	767	1.0	1.000	403	8	767	1.0	564	11	1074	1.4	134	3	256	0	269	5	511	0.7	
U- 11	Canister and Decon Survey	389	7	5	1.0	1.000	389	7	5	1.0	545	10	7	1.4	130	2	2	0	259	5	3	0.7	
U- 12	Canister Loading		7		0.5	1.000	0	7	0	0.5	0	10	0	0.7	0	2	0	0	0	5	0	0.3	
U- 13	Hardware and Scrap Metal Pkg.	119	1	10	0.5	1.000	119	1	10	0.5	167	1	14	0.7	40	0	3	0	79	1	7	0.3	
U- 14	Util & Service Air / Water	112	7		0.5	1.000	112	7	0	0.5	157	10	0	0.7	37	2	0	0	75	5	0	0.3	
U- 15	Emergency Power	77	3			1.000	77	3	0	0.0	108	4	0	0.0	26	1	0	0	51	2	0	0.0	
U- 16	Heating Ventilating	165	51	100		1.000	165	51	100	0.0	231	71	140	0.0	55	17	33	0	110	34	67	0.0	
U- 17	Radiation Monitoring	186	12		1.0	1.000	186	12	0	1.0	260	17	0	1.4	62	4	0	0	124	8	0	0.7	
U- 18	Electrical & Controls	175	23			1.000	175	23	0	0.0	245	32	0	0.0	58	8	0	0	117	15	0	0.0	
U- 19	Maintenance Hot Cell	504	6			1.000	504	6	0	0.0	706	8	0	0.0	168	2	0	0	336	4	0	0.0	
U- 20	Manipulator Maintenance	200	6		0.3	1.000	200	6	0	0.3	280	8	0	0.4	67	2	0	0	133	4	0	0.2	
U- 21	Fresh Container Del & Stor	91	7			1.000	91	7	0	0.0	127	10	0	0.0	30	2	0	0	61	5	0	0.0	
U- 22	Others	11	1			1.000	11	1	0	0.0	15	1	0	0.0	4	0	0	0	7	1	0	0.0	
Total (1993 Dollars)		7299	210	1078	1442		5678	197	1022	1257	8229	278	1430	1759	1959	66	341	419	3918	131	681	836	
			FTE(K\$)	\$140		Grand Total				8353 Grand Total				11694 Grand Total				2784 Grand Total				5569	

Note: Costs above are in 1993 Dollars. Convert to 1996 dollars multiplying by 1.098.

93-94	94-95	95-96	Cum	6454	216	1122	1380	9035	303	1570	1932	2151	72	374	460	4302	144	748	920
102.60%	103.10%	103.80%	109.80%	Grand Total (\$1996)				9172 Grand Total (\$1996)	12640 Grand Total (\$1996)				3057 Grand Total (\$1996)				6114		

**Table C8.1-1 Transfer Facility Estimate - Direct Disposal
Sheet 4**

Decontamination and Decommissioning Costs

	Area Description	EGG-WM	Scaling		Transfer
		10708	Cap	Scale	Facility
		D&D	Ratio	Factor	D&D
U- 1	Receive and Inspect	972		1.000	972
U- 2	Wash Vehicle and Cask	4050		1.000	4050
U- 3	Cask Staging and Handling	919		1.000	919
U- 4	SNF Loading	368		1.000	368
U- 5	De-Canning	368	0.500	0.660	243
U- 6	Sampling	511	0.000	0.000	0
U- 7	SNF Analysis	1823	0.300	0.486	885
U- 8	Waste Treatment and Pkg	459	0.600	0.736	338
U- 9	Conditioning	950	0.600	0.736	699
U- 10	Canning	1072		1.000	1072
U- 11	Canister and Decon Survey	919		1.000	919
U- 12	Canister Loading	460		1.000	460
U- 13	Hardware and Scrap Metal Pkg.	540		1.000	540
U- 14	Util & Service Air / Water	540		1.000	540
U- 15	Emergency Power	1215		1.000	1215
U- 16	Heating Ventilating	405		1.000	405
U- 17	Radiation Monitoring	540		1.000	540
U- 18	Electrical & Controls	1563		1.000	1563
U- 19	Maintenance Hot Cell	540		1.000	540
U- 20	Manipulator Maintenance	608		1.000	608
U- 21	Fresh Container Del & Stor			1.000	0
U- 22	Others			1.000	0
Total (1993 Dollars)		18822			16876

Note: Costs above are in 1993 Dollars. Convert to 1998 dollars multiplying by 1.098. ----->>>

18530

93-94 94-95 95-96 Cum
102.60% 103.10% 103.80% 109.80%

Table C8.1-2 Transfer Facility Estimate Based On OCRWM MRS

CI Description	1994 MPC CDR Estimate (\$1992)	Converted MPC CDR Estimate (\$1996)	Capacity Ratio (C2/C1)	Cost Ratio (C2/C1)**R (R=.6)	Estimated Scale Factor (%)	SRS Transfer Facility Estimate (\$1996)
		111.8%				
Site Development	\$25,806,201	\$28,845,311		0.10	10%	\$2,884,531
Transfer Facility	\$59,508,088	\$66,516,158		0.50	50%	\$33,258,079
Storage Mode Facility	\$12,248,474	\$13,690,936	0.00	0.00	0%	\$0
Radwaste Facility	\$7,347,592	\$8,212,893	0.70	0.81	70%	\$6,630,633
Security Facilities	\$10,686,735	\$11,945,276	0.00	0.00	0%	\$0
Fire Protection	\$1,387,845	\$1,551,287	0.20	0.38	40%	\$590,623
Compressed Air Services	\$666,746	\$745,266	0.40	0.58	60%	\$430,078
Electric Power Delivery	\$9,879,917	\$11,043,442	0.20	0.38	30%	\$4,204,578
Instrumentation & Controls	\$15,091,290	\$16,868,541	0.60	0.74	70%	\$12,415,616
Communications	\$775,647	\$866,992	0.70	0.81	80%	\$699,961
Administration and Site Services	\$5,341,086	\$5,970,088	0.00	0.00	0%	\$0
Site Services Warehouse	\$844,534	\$943,992	0.00	0.00	0%	\$0
Utility Warehouse	\$1,351,876	\$1,511,082	0.00	0.00	0%	\$0
Protected Area Warehouse	\$451,405	\$504,565	0.00	0.00	0%	\$0
Vehicle Maintenance Facility	\$1,363,890	\$1,524,511	0.00	0.00	0%	\$0
Fossil Fuel Distribution	\$1,673,625	\$1,870,722	0.00	0.00	0%	\$0
Site Vehicles	\$2,611,609	\$2,919,170	0.00	0.00	0%	\$0
Water Utilities Facility	\$4,622,330	\$5,166,686	0.00	0.00	0%	\$0
Conventional Waste Treatment	\$1,160,415	\$1,297,073	0.20	0.38	20%	\$493,836
Sanitary Waste Treatment	\$278,052	\$310,797	0.20	0.38	30%	\$118,330
Visitors and Media Center	\$2,880,295	\$3,219,498	0.00	0.00	0%	\$0
Cask Maintenance Facility	\$18,591,742	\$20,781,230	0.00	0.00	0%	\$0
Direct Cost Subtotal	\$184,569,394	\$206,305,518				\$61,726,266

Table C8.1-3

Total System Cost (\$million) Direct Disposal (1100 Small Packages)

	Cost Items	R&D / Development	Facility & Equipment	Operations	D & D	Total
A	WET STORAGE AND HANDLING - Wet Transfer and Storage	0.0	0.0	282.2 282.2	0.0	282.2
B	TRANSFER AND PACKAGING - Transfer Facility - Canisters	10.0 10.0	184.6 169.6 15.0	226.7 184.0 42.7	18.5 18.5	439.8
C	TREATMENT	0.0	0.0	0.0	0.0	0.0
D	INTERIM STORAGE - Vault Storage	0.0	56.0 56.0	61.8 61.8	2.3 2.3	120.1
E	DISPOSAL E1 - Transportation E2 - Repository Fee E3 - Repository Operations	30.0 30.0	10.0 10.0	394.8 26.7 368.1	0.0	434.8
F	ADJUSTMENTS - Hot Vacuum Drying Facility	0.0	31.2 31.2	25.0 25.0	5.0 5.0	61.2
	TOTALS	40.0	281.8	990.4	25.9	1338.1

Table C8.1-4 Annual Cost Summary -- Time Phase DD

Year	1996	1997	1998	1999	2000	2001	2002	2003	2004	2005	2006	2007	2008	2009	2010	2011	2012	2013	2014	2015	2016	2017	2018	2019	2020	
Wet Storage and Transfer																										
RBOF Operations	15.0	15.0	15.0	15.0	15.0	8.2	8.2	8.2	8.2	8.2	8.2															
L-Basin Operations	17.2	17.2	17.2	17.2	17.2	12.0	12.0	12.0	12.0	12.0	12.0															
Subtotal	32.2	32.2	32.2	32.2	32.2	20.2	20.2	20.2	20.2	20.2	20.2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Dry Transfer Facility																										
Development		2.0	2.0	4.0	2.0																					
Implementation			39.3	72.9	57.4																					
Operations						12.8	12.8	12.8	12.8	12.8	12.8	6.1	6.1	6.1	6.1	6.1	6.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1
Decontam & Decomm																										
Subtotal	0.0	2.0	41.3	76.9	59.4	12.8	12.8	12.8	12.8	12.8	12.8	6.1	6.1	6.1	6.1	6.1	6.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1
Dry Storage Facility																										
Development																										
Implementation																										
Operations			13.0	24.1	19.0	3.2	3.2	3.2	3.2	3.2	3.2	1.8	1.8	1.8	1.8	1.8	1.8	1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.8
Decontam & Decomm																										
Subtotal	0.0	0.0	13.0	24.1	19.0	3.2	3.2	3.2	3.2	3.2	3.2	1.8	1.8	1.8	1.8	1.8	1.8	1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.8
Dry Storage Canisters																										
Development																										
Implementation																										
Operations			5.0	5.0	5.0	4.3	4.3	4.3	4.3	4.3	4.3	2.1	2.1	2.1	2.1	2.1	2.1	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2
Decontam & Decomm																										
Subtotal	0.0	0.0	5.0	5.0	5.0	4.3	4.3	4.3	4.3	4.3	4.3	2.1	2.1	2.1	2.1	2.1	2.1	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2
Hot Vacuum Drying Facility																										
Development																										
Implementation				1.5	8.0	12.6	9.1																			
Operations							5.0	10.0	10.0																	
Decontam & Decomm																										
Subtotal	0.0	0.0	0.0	1.5	8.0	12.6	14.1	10.0	10.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Transportation System																										
Development																										
Implementation					2.0	2.0	2.0																		4.0	2.7
Operations																										
Decontam & Decomm																										
Subtotal	0.0	0.0	0.0	0.0	2.0	2.0	2.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4.0	2.7
Disposal (Repository)																										
Development (Fee)																										
Implementation																										
Operations																										
Decontam & Decomm																										
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.0	5.0	5.0	5.0	5.0	5.0	0.0	36.8
System Total																										
Development	0.0	2.0	2.0	4.0	2.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.0	5.0	5.0	5.0	5.0	5.0	0.0	0.0
Implementation	0.0	0.0	57.3	103.4	91.3	14.6	11.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4.0	0.0
Operations	32.2	32.2	32.2	32.2	32.2	40.5	45.5	50.5	50.5	40.5	40.5	10.0	10.0	10.0	10.0	10.0	10.0	4.4	4.4	4.4	4.4	4.4	4.4	4.4	4.4	44.5
Decontam & Decomm	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Subtotal	32.2	34.2	91.5	139.6	125.5	55.1	56.6	50.5	50.5	40.5	40.5	10.0	10.0	10.0	10.0	10.0	10.0	9.4	9.4	9.4	9.4	9.4	9.4	9.4	8.4	44.5

Table C8.1-4 Annual Cost Summary -- Time Phase DD

Year	2021	2022	2023	2024	2025	2026	2027	2028	2029	2030	2031	2032	2033	2034	2035	2036	2037	Total
Wet Storage and Transfer																		
RBOF Operations																		124.2
L-Basin Operations																		158.0
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	282.2
Dry Transfer Facility																		
Development																		10.0
Implementation																		169.6
Operations	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1			184.0
Decontam & Decomm																9.3	9.3	18.5
Subtotal	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	9.3	9.3	382.1
Dry Storage Facility																		
Development																		0.0
Implementation																		56.0
Operations	1.8	1.8	1.8	1.8	1.8	1.8	1.8	1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.2			61.8
Decontam & Decomm																1.2	1.2	2.3
Subtotal	1.8	1.8	1.8	1.8	1.8	1.8	1.8	1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.2	120.1
Dry Storage Canisters																		
Development																		0.0
Implementation																		15.0
Operations	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2		42.7
Decontam & Decomm																		0.0
Subtotal	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.0	57.7
Hot Vacuum Drying Facility																		
Development																		0.0
Implementation																		31.2
Operations																		25.0
Decontam & Decomm																2.5	2.5	5.0
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.5	2.5	61.2
Transportation System																		
Development																		0.0
Implementation																		10.0
Operations	2.7	2.7	2.7	2.7	2.7	2.7	2.7	0.7	0.7	0.7	0.7	0.7	0.7	0.7	0.7			26.7
Decontam & Decomm																		0.0
Subtotal	2.7	2.7	2.7	2.7	2.7	2.7	2.7	0.7	0.7	0.7	0.7	0.7	0.7	0.7	0.7	0.0	0.0	36.7
Disposal (Repository)																		
Development (Fee)																		30.0
Implementation																		0.0
Operations	36.8	36.8	36.8	36.8	36.8	36.8	36.8	9.2	9.2	9.2	9.2	9.2	9.2	9.2	9.2			368.1
Decontam & Decomm																		0.0
Subtotal	36.8	36.8	36.8	36.8	36.8	36.8	36.8	9.2	9.2	9.2	9.2	9.2	9.2	9.2	9.2	0.0	0.0	398.1
System Total																		
Development	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	40.0
Implementation	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	281.8
Operations	44.5	44.5	44.5	44.5	44.5	44.5	44.5	14.3	14.3	14.3	14.3	14.3	14.3	14.3	14.3	0.0	0.0	990.4
Decontam & Decomm	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	12.9	12.9	25.9
Subtotal	44.5	44.5	44.5	44.5	44.5	44.5	44.5	14.3	14.3	14.3	14.3	14.3	14.3	14.3	14.3	12.9	12.9	1338.1

8.2 Co-Disposal

The Transfer Facility for the Co-Disposal Option is the same as that for the Direct Disposal Option described in Section 8.1. Table C8.2-1 shows a roll up of the total system costs, and Table C8.2-2 presents total system costs and their distribution in time.

**Table C8.2-1
Total System Cost Direct Disposal - CoDisposal (1400 Co-Disposal Pkgs)**

	Cost Items	R&D / Development	Facility & Equipment	Operations	D & D	Total
A	WET STORAGE AND HANDLING - Wet Transfer and Storage	0.0	0.0	282.2 282.2	0.0	282.2
B	TRANSFER AND PACKAGING - Transfer Facility - Canisters	10.0 10.0	184.6 169.6 15.0	213.4 184.0 29.4	18.5 18.5	426.5
C	TREATMENT	0.0	0.0	0.0	0.0	0.0
D	INTERIM STORAGE - Vault Storage	0.0	60.1 60.1	63.4 63.4	2.6 2.6	126.0
E	DISPOSAL E1 - Transportation E2 - Repository Fee E3 - Repository Operations	30.0 30.0	10.0 10.0	172.0 32.0 140.0	0.0	212.0
F	CREDITS - Hot Vacuum Drying Facility	0.0	31.2 31.2	25.0 25.0	5.0 5.0	61.2
	TOTALS	40.0	285.8	756.0	26.1	1107.9

Table C8.2-2 Annual Cost Summary -- Time Phase DD - CoD

Year	1996	1997	1998	1999	2000	2001	2002	2003	2004	2005	2006	2007	2008	2009	2010	2011	2012	2013	2014	2015	2016	2017	2018	2019	2020	
Wet Storage and Transfer																										
RBOF Operations	15.0	15.0	15.0	15.0	15.0	8.2	8.2	8.2	8.2	8.2	8.2															
L-Basin Operations	17.2	17.2	17.2	17.2	17.2	12.0	12.0	12.0	12.0	12.0	12.0															
Subtotal	32.2	32.2	32.2	32.2	32.2	20.2	20.2	20.2	20.2	20.2	20.2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Dry Transfer Facility																										
Development		2.0	2.0	4.0	2.0																					
Implementation			39.3	72.9	57.4																					
Operations						12.8	12.8	12.8	12.8	12.8	12.8	6.1	6.1	6.1	6.1	6.1	6.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1
Decontam & Decomm																										
Subtotal	0.0	2.0	41.3	76.9	59.4	12.8	12.8	12.8	12.8	12.8	12.8	6.1	6.1	6.1	6.1	6.1	6.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1
Dry Storage Facility																										
Development																										
Implementation			13.9	25.8	20.3																					
Operations						3.3	3.3	3.3	3.3	3.3	3.3	1.8	1.8	1.8	1.8	1.8	1.8	1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.8
Decontam & Decomm																										
Subtotal	0.0	0.0	13.9	25.8	20.3	3.3	3.3	3.3	3.3	3.3	3.3	1.8	1.8	1.8	1.8	1.8	1.8	1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.8
Dry Storage Canisters																										
Development																										
Implementation			5.0	5.0	5.0																					
Operations						2.9	2.9	2.9	2.9	2.9	2.9	1.5	1.5	1.5	1.5	1.5	1.5	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1
Decontam & Decomm																										
Subtotal	0.0	0.0	5.0	5.0	5.0	2.9	2.9	2.9	2.9	2.9	2.9	1.5	1.5	1.5	1.5	1.5	1.5	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1
Hot Vacuum Drying Facility																										
Development																										
Implementation				1.5	8.0	12.6																				
Operations						9.1	5.0	10.0	10.0																	
Decontam & Decomm																										
Subtotal	0.0	0.0	0.0	1.5	8.0	12.6	14.1	10.0	10.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Transportation System																										
Development																										
Implementation					2.0	2.0																				
Operations																									4.0	3.2
Decontam & Decomm																										
Subtotal	0.0	0.0	0.0	0.0	2.0	2.0	2.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4.0	3.2
Disposal (Repository)																										
Development (Fee)																										
Implementation																										
Operations																										
Decontam & Decomm																										
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.0	5.0	5.0	5.0	5.0	5.0	0.0	14.0
System Total																										
Development	0.0	2.0	2.0	4.0	2.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.0	5.0	5.0	5.0	5.0	5.0	0.0	0.0
Implementation	0.0	0.0	58.2	105.2	92.7	14.6	11.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4.0	0.0
Operations	32.2	32.2	32.2	32.2	32.2	39.3	44.3	49.3	49.3	39.3	39.3	9.4	9.4	9.4	9.4	9.4	9.4	4.4	4.4	4.4	4.4	4.4	4.4	4.4	4.4	22.2
Decontam & Decomm	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Subtotal	32.2	34.2	92.4	141.4	126.9	53.9	55.4	49.3	49.3	39.3	39.3	9.4	9.4	9.4	9.4	9.4	9.4	9.4	9.4	9.4	9.4	9.4	9.4	9.4	8.4	22.2

Table C8.2-2 Annual Cost Summary -- Time Phase DD - CoD

Year	2021	2022	2023	2024	2025	2026	2027	2028	2029	2030	2031	2032	2033	2034	2035	2036	2037	Total
Wet Storage and Transfer																		
RBOF Operations																		124.2
L-Basin Operations																		158.0
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	282.2
Dry Transfer Facility																		
Development																		10.0
Implementation																		169.6
Operations	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1			184.0
Decontam & Decomm																9.3	9.3	18.5
Subtotal	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	9.3	9.3	382.1
Dry Storage Facility																		
Development																		0.0
Implementation																		60.1
Operations	1.8	1.8	1.8	1.8	1.8	1.8	1.8	1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.2			63.4
Decontam & Decomm																1.3	1.3	2.6
Subtotal	1.8	1.8	1.8	1.8	1.8	1.8	1.8	1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.3	1.3	126.0
Dry Storage Canisters																		
Development																		0.0
Implementation																		15.0
Operations	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1			29.4
Decontam & Decomm																		0.0
Subtotal	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.0	0.0	44.4
Hot Vacuum Drying Facility																		
Development																		0.0
Implementation																		31.2
Operations																		25.0
Decontam & Decomm																2.5	2.5	5.0
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.5	2.5	61.2
Transportation System																		
Development																		0.0
Implementation																		10.0
Operations	3.2	3.2	3.2	3.2	3.2	3.2	3.2	0.8	0.8	0.8	0.8	0.8	0.8	0.8	0.8			32.0
Decontam & Decomm																		0.0
Subtotal	3.2	3.2	3.2	3.2	3.2	3.2	3.2	0.8	0.8	0.8	0.8	0.8	0.8	0.8	0.8	0.0	0.0	42.0
Disposal (Repository)																		
Development (Fee)																		30.0
Implementation																		0.0
Operations	14.0	14.0	14.0	14.0	14.0	14.0	14.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5			140.0
Decontam & Decomm																		0.0
Subtotal	14.0	14.0	14.0	14.0	14.0	14.0	14.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	0.0	0.0	170.0
System Total																		
Development	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	40.0
Implementation	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	285.8
Operations	22.2	22.2	22.2	22.2	22.2	22.2	22.2	8.7	8.7	8.7	8.7	8.7	8.7	8.7	8.7	0.0	0.0	756.0
Decontam & Decomm	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	13.1	13.1	26.1
Subtotal	22.2	22.2	22.2	22.2	22.2	22.2	22.2	8.7	8.7	8.7	8.7	8.7	8.7	8.7	8.7	13.1	13.1	1107.9

8.3 Press and Dilute (20%)

The first variation of this treatment technology considers a process in which the SNF is diluted to less than 20% enrichment. In addition to the functions of the dry transfer facility, a press and dilute module is conceptualized to perform the functions associated with this technology.

Table C8.3-1 shows the estimated cost for additional equipment for the Press and Dilute module.

Table C8.3-1 Additional Equipment for Press and Dilute Option

<u>Equipment</u>	<u>Cost (Thousands of 1996 dollars)</u>
Press & Gantry Crane	1,367
Misc. Equipment	4,479
Allowances	2,770
Total Press & Dilute	8,616

Additional building space for this treatment module is 1600 ft² of hot cell space, costed at \$2306 per square foot (1996 dollars), consistent with the transfer facility basis, for an additional building cost of \$3,689,000.

Additional operating and maintenance costs were estimated based on WSRC-TR-95-0180 Life Cycle Cost Estimates for Disposal of Aluminum-Clad HEU SNF (U), scaled to the same value as the melt and dilute option in that study based on higher throughput. The additional annual costs for a 15-shift workweek are shown in Table C8.3-2. Costs for other work patterns were scaled from this value.

Table C8.3-2 Additional Operating and Maintenance Costs for Press and Dilute Option

<u>Item</u>	<u>Cost (Thousands of 1996 dollars)</u>
Labor	1,204
Utilities	33
Materials	5,126
Maintenance Contracts	1,688

Incremental additional cost for decontamination and decommissioning of the Press and Dilute module is estimated to be \$4 million.

These costs are added to those of the transfer facility for the direct disposal options. Table C8.3-3 presents a summary of the total project costs for the transfer facility with the treatment module included, including design, construction, operation and decommissioning, including appropriate funding overheads. Table C8.3-4 summarizes overall costs for this option, and Table C8.3-5 shows the distribution of these costs in time.

**Table C8.3-4
Total System Cost (\$Millions) Press and Dilute (400 Co-Disposal Pkgs - 20%)**

	Cost Items	R&D / Development	Facility & Equipment	Operations	D & D	Total
A	WET STORAGE AND HANDLING - Wet Transfer and Storage	0.0	0.0	346.6 346.6	0.0	346.6
B	TRANSFER AND PACKAGING - Treatment Facility (Recv and Pkg) - Canisters	0.0	184.6 169.6 15.0	213.8 205.4 8.4	18.5 18.5	416.9
C	TREATMENT - Treatment Facility (Treatment Module)	15.0 15.0	31.4 31.4	180.3 180.3	4.0 4.0	230.7
D	INTERIM STORAGE - Vault Storage	0.0	41.1 41.1	54.4 54.4	1.4 1.4	96.9
E	DISPOSAL E1 - Transportation E2 - Repository Fee E3 - Repository Operations	30.0 30.0	10.0 10.0	49.1 9.1 40.0	0.0	89.1
F	ADJUSTMENTS - Hot Vacuum Drying Facility	0.0	31.2 31.2	25.0 25.0	5.0 5.0	61.2
	TOTALS	45.0	298.3	869.3	29.0	1241.5

Table C8.3-3 Receipt and Treatment Facility - Press & Dilute

Item	Description	Estimated Cost
DEVELOPMENT COSTS		
0.0	Development	
0.1	Research and Development	7,500
0.2	Process Demonstration	7,500
0.3	Waste Form Qualification	0
0.4	Development Contingency (0% of 0.1, 0.2, and 0.3)	
	SUBTOTAL 0.0 DEVELOPMENT	15,000

IMPLEMENTATION COSTS		
1.0	Construction Costs (Fixed Price Contract)	
1.1	Building Structure Costs	40,088
1.2	Building Equipment Costs	38,754
1.3	Construction Indirects (29% of 1.1 and 1.2)	22,864
	SUBTOTAL Construction 1.0	101,706
2.0	Title I and Title II Design (8% of 1.0)	8,136
3.0	Inspection (3% of 1.0)	3,051
4.0	Project Support (See Note 1) (6% of 1.0)	6,102
5.0	Construction Management (14% of 1.0)	14,239
6.0	SUBTOTAL Engr & Const (1.0, 2.0, 3.0, 4.0, 5.0)	133,235
7.0	Site G & A (16% of 6.0)	21,318
8.0	Total Estimated Cost (TEC) (6.0, 7.0)	154,553
9.0	OPC (Other Project Costs) (See Note 2) (30% of 8.0)	46,366
10.0	Implementation Contingency (0% of 8.0 and 9.0)	
11.0	Total Projected Cost (8.0, 9.0, 10.0) IMPLEMENTATION	200,919

OPERATIONS COSTS					
		5 Shifts / Week	10 Shifts / Week	15 Shifts / Week	21 Shifts / Week
12.0	Operations and Maintenance Costs (See Note 3)				
12.1	Annual Operation Costs (Labor)	861	1723	2584	3617
12.2	Annual Utility Costs	83	166	249	349
12.3	Annual Material / Consumable Costs	2,083	4165	6248	8747
12.4	Annual Maintenance Costs	2,714	5428	8142	11398
12.5	Interim Storage Costs				
12.6	Operations Contingency (0% of 12.1 through 12.5)				
	SUBTOTAL 12.0 OPERATIONS (12.1 through 12.6)	5,741	11,482	17,223	24,112

DECONTAMINATION AND DECOMMISSIONING COSTS		
13.0	Decontamination and Decommissioning Costs	
13.1	Facility and Equipment D&D	22,530
13.2	Storage D&D	
	SUBTOTAL 13.0 D&D	22,530

NOTES:

1. Project Support includes Project Management, Project Control, Estimating and QA
2. Other Project Costs include NEPA process, developing operating procedures, startup and checkout.
3. Annual costs may vary with years depending on operations performed.
4. Percentages shown above are suggested starting point values, and may be adjusted.

Table C8.3-5 Sheet1 Annual Cost Summary -- Time Phase Press & Dilute

Year	1996	1997	1998	1999	2000	2001	2002	2003	2004	2005	2006	2007	2008	2009	2010	2011	2012	2013	2014	2015	2016	2017
Wet Storage and Transfer																						
RBOF Operations	15.0	15.0	15.0	15.0	15.0	15.0	15.0	8.2	8.2	8.2	8.2	8.2	8.2									
L-Basin Operations	17.2	17.2	17.2	17.2	17.2	17.2	17.2	12.0	12.0	12.0	12.0	12.0	12.0									
Subtotal	32.2	32.2	32.2	32.2	32.2	32.2	32.2	20.2	20.2	20.2	20.2	20.2	20.2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Receipt and Treatment Facility																						
Development		2.3	2.3	4.5	4.5	1.5																
Implementation				7.1	54.4	81.6	57.8															
Operations								24.1	24.1	24.1	24.1	24.1	24.1	11.5	11.5	11.5	11.5	11.5	11.5	11.5	5.7	5.7
Decontam & Decomm																						
Subtotal	0.0	2.3	2.3	11.6	58.9	83.1	57.8	24.1	24.1	24.1	24.1	24.1	24.1	11.5	11.5	11.5	11.5	11.5	11.5	11.5	5.7	5.7
Dry Storage Facility																						
Development																						
Implementation				1.4	11.1	16.7	11.8															
Operations								2.9	2.9	2.9	2.9	2.9	2.9	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.0
Decontam & Decomm																						
Subtotal	0.0	0.0	0.0	1.4	11.1	16.7	11.8	2.9	2.9	2.9	2.9	2.9	2.9	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.0
Hot Vacuum Drying Facility																						
Development																						
Implementation						1.5	8.0	12.6	9.1													
Operations									5.0	10.0	10.0											
Decontam & Decomm																						
Subtotal	0.0	0.0	0.0	0.0	0.0	1.5	8.0	12.6	14.1	10.0	10.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Co-Disposal Canisters																						
Development																						
Implementation					5.0	5.0	5.0															
Operations								0.8	0.8	0.8	0.8	0.8	0.8	0.3	0.3	0.3	0.3	0.3	0.3	0.3	0.3	0.1
Decontam & Decomm																						
Subtotal	0.0	0.0	0.0	0.0	5.0	5.0	5.0	0.8	0.8	0.8	0.8	0.8	0.8	0.3	0.3	0.3	0.3	0.3	0.3	0.3	0.3	0.1
Transportation System																						
Development																						
Implementation					2.0	2.0	2.0															
Operations																						
Decontam & Decomm																						
Subtotal	0.0	0.0	0.0	0.0	2.0	2.0	2.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Disposal (Repository)																						
Development (Fee)																						
Implementation																					5.0	5.0
Operations																						
Decontam & Decomm																						
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.0	5.0	5.0
System Total																						
Development	0.0	2.3	2.3	4.5	4.5	1.5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.0	5.0	5.0
Implementation	0.0	0.0	0.0	8.5	72.6	106.8	84.7	12.6	9.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Operations	32.2	32.2	32.2	32.2	32.2	32.2	32.2	48.1	53.1	58.1	58.1	48.1	48.1	13.4	13.4	13.4	13.4	13.4	13.4	13.4	6.8	6.8
Decontam & Decomm	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Subtotal	32.2	34.5	34.5	45.2	109.3	140.5	116.9	60.7	62.2	58.1	58.1	48.1	48.1	13.4	13.4	13.4	13.4	18.4	18.4	18.4	11.8	11.8

Table C8.3-5 Sheet2 Annual Cost Summary -- Time Phase Press & Dilute

Year	2018	2019	2020	2021	2022	2023	2024	2025	2026	2027	2028	2029	2030	2031	2032	2033	2034	2035	2036	2037	Total	
Wet Storage and Transfer																						
RBOF Operations																					154.2	
L-Basin Operations																						192.4
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	346.6	
Receipt and Treatment Facility																						
Development																					15.0	
Implementation																					200.9	
Operations	5.7	5.7	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	5.7	5.7	5.7	5.7	5.7	5.7	5.7	5.7			385.8	
Decontam & Decomm																			11.3	11.3	22.5	
Subtotal	5.7	5.7	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	5.7	5.7	5.7	5.7	5.7	5.7	5.7	5.7	11.3	11.3	624.2	
Dry Storage Facility																						
Development																					0.0	
Implementation																					41.1	
Operations	1.0	1.0	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0			54.4	
Decontam & Decomm																			0.7	0.7	1.4	
Subtotal	1.0	1.0	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	0.7	0.7	96.9	
Hot Vacuum Drying Facility																						
Development																					0.0	
Implementation																					31.2	
Operations																					25.0	
Decontam & Decomm																			2.5	2.5	5.0	
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.5	2.5	61.2	
Co-Disposal Canisters																						
Development																					0.0	
Implementation																					15.0	
Operations	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1		8.4	
Decontam & Decomm																					0.0	
Subtotal	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.0	23.4	
Transportation System																						
Development																					0.0	
Implementation		4.0																			10.0	
Operations			0.9	0.9	0.9	0.9	0.9	0.9	0.9	0.9	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2		9.1	
Decontam & Decomm																					0.0	
Subtotal	0.0	4.0	0.9	0.9	0.9	0.9	0.9	0.9	0.9	0.9	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.0	19.1	
Disposal (Repository)																						
Development (Fee)	5.0																				30.0	
Implementation																					0.0	
Operations			4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0			40.0	
Decontam & Decomm																					0.0	
Subtotal	5.0	0.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	0.0	0.0	70.0	
System Total																						
Development	5.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	45.0	
Implementation	0.0	4.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	298.3	
Operations	6.8	6.8	18.1	18.1	18.1	18.1	18.1	18.1	18.1	18.1	8.1	8.1	8.1	8.1	8.1	8.1	8.1	8.1	8.1	0.0	869.3	
Decontam & Decomm	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	14.5	14.5	29.0	
Subtotal	11.8	10.8	18.1	18.1	18.1	18.1	18.1	18.1	18.1	18.1	8.1	8.1	8.1	8.1	8.1	8.1	8.1	8.1	14.5	14.5	1241.5	

8.4 Press and Dilute (2%)

An additional variation of the press and dilute treatment technology considers a process in which the SNF is further diluted to less than 2% enrichment. The increased facility requirements for this option are the same as the first press and dilute option. Based on previous experience, the transfer facility handling of transportation casks will be the limiting factor for the transfer operation, and the treatment module in this estimate will be adequate for the additional material throughput for increased volume of dilution materials. The significant difference between the two press and dilute options is in those portions of the estimate which scale on the number of packages. This includes the canister costs, transportation, and disposal costs. Table C8.4-1 summarizes overall costs for the Press and Dilute (2%) option, and Table C8.4-2 shows the distribution of these costs in time.

Table C8.4-1
Total System Cost (\$million) Press and Dilute (1300 Co-Disposal Pkgs - 2%)

	Cost Items	R&D/ Development	Facility & Equipment	Operations	D & D	Total
A	WET STORAGE AND HANDLING - Wet Transfer and Storage	0.0	0.0	346.6 346.6	0.0	346.6
B	TRANSFER AND PACKAGING - Treatment Facility (Recv and Pkg) - Canisters	0.0	184.6 169.6 15.0	232.7 205.4 27.3	18.5 18.5	435.8
C	TREATMENT - Treatment Facility (Treatment Module)	15.0 15.0	31.4 31.4	180.3 180.3	4.0 4.0	230.7
D	INTERIM STORAGE - Vault Storage	0.0	58.5 58.5	61.1 61.1	2.5 2.5	122.1
E	DISPOSAL E1 - Transportation E2 - Repository Fee E3 - Repository Operations	30.0 30.0	10.0 10.0	159.7 29.7 130.0	0.0	199.7
F	ADJUSTMENTS - Hot Vacuum Drying Facility	0.0	31.2 31.2	25.0 25.0	5.0 5.0	61.2
	TOTALS	45.0	315.6	1005.5	30.0	1396.1

Table C8.4-2 Annual Cost Summary -- Time Phase Press & Dilute (2%)

Year	1996	1997	1998	1999	2000	2001	2002	2003	2004	2005	2006	2007	2008	2009	2010	2011	2012	2013	2014	2015	2016	2017	
Wet Storage and Transfer																							
RBOF Operations	15.0	15.0	15.0	15.0	15.0	15.0	15.0	8.2	8.2	8.2	8.2	8.2	8.2										
L-Basin Operations	17.2	17.2	17.2	17.2	17.2	17.2	17.2	12.0	12.0	12.0	12.0	12.0	12.0										
Subtotal	32.2	32.2	32.2	32.2	32.2	32.2	32.2	20.2	20.2	20.2	20.2	20.2	20.2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	
Receipt and Treatment Facility																							
Development		2.3	2.3	4.5	4.5	1.5																	
Implementation				7.1	54.4	81.6	57.8																
Operations								24.1	24.1	24.1	24.1	24.1	24.1	11.5	11.5	11.5	11.5	11.5	11.5	11.5	5.7	5.7	
Decontam & Decomm																							
Subtotal	0.0	2.3	2.3	11.6	58.9	83.1	57.8	24.1	24.1	24.1	24.1	24.1	24.1	11.5	11.5	11.5	11.5	11.5	11.5	11.5	5.7	5.7	
Dry Storage Facility																							
Development																							
Implementation				2.1	15.9	23.8	16.8																
Operations								3.3	3.3	3.3	3.3	3.3	3.3	1.8	1.8	1.8	1.8	1.8	1.8	1.8	1.2	1.2	
Decontam & Decomm																							
Subtotal	0.0	0.0	0.0	2.1	15.9	23.8	16.8	3.3	3.3	3.3	3.3	3.3	3.3	1.8	1.8	1.8	1.8	1.8	1.8	1.8	1.2	1.2	
Hot Vacuum Drying Facility																							
Development																							
Implementation						1.5	8.0	12.6	9.1														
Operations									5.0	10.0	10.0												
Decontam & Decomm																							
Subtotal	0.0	0.0	0.0	0.0	0.0	1.5	8.0	12.6	14.1	10.0	10.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	
Co-Disposal Canisters																							
Development																							
Implementation					5.0	5.0	5.0																
Operations								2.7	2.7	2.7	2.7	2.7	2.7	1.1	1.1	1.1	1.1	1.1	1.1	1.1	0.2	0.2	
Decontam & Decomm																							
Subtotal	0.0	0.0	0.0	0.0	5.0	5.0	5.0	2.7	2.7	2.7	2.7	2.7	2.7	1.1	1.1	1.1	1.1	1.1	1.1	1.1	0.2	0.2	
Transportation System																							
Development																							
Implementation					2.0	2.0	2.0																
Operations																							
Decontam & Decomm																							
Subtotal	0.0	0.0	0.0	0.0	2.0	2.0	2.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	
Disposal (Repository)																							
Development (Fee)																				5.0	5.0	5.0	5.0
Implementation																							
Operations																							
Decontam & Decomm																							
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.0	5.0	5.0	5.0	
System Total																							
Development	0.0	2.3	2.3	4.5	4.5	1.5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.0	5.0	5.0	5.0	
Implementation	0.0	0.0	0.0	9.1	77.3	113.8	89.7	12.6	9.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	
Operations	32.2	32.2	32.2	32.2	32.2	32.2	32.2	50.3	55.3	60.3	60.3	50.3	50.3	14.4	14.4	14.4	14.4	14.4	14.4	14.4	7.1	7.1	
Decontam & Decomm	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	
Subtotal	32.2	34.5	34.5	45.8	114.0	147.5	121.9	62.9	64.4	60.3	60.3	50.3	50.3	14.4	14.4	14.4	14.4	14.4	19.4	19.4	19.4	12.1	

Table C8.4-2 Annual Cost Summary -- Time Phase Press & Dilute (2%)

Year	2018	2019	2020	2021	2022	2023	2024	2025	2026	2027	2028	2029	2030	2031	2032	2033	2034	2035	2036	2037	Total
Wet Storage and Transfer																					
RBOF Operations																					154.2
L-Basin Operations																					192.4
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	346.6
Receipt and Treatment Facility																					
Development																					15.0
Implementation																					200.9
Operations	5.7	5.7	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	5.7	5.7	5.7	5.7	5.7	5.7	5.7	5.7			385.8
Decontam & Decomm																			11.3	11.3	22.5
Subtotal	5.7	5.7	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	5.7	5.7	5.7	5.7	5.7	5.7	5.7	5.7	11.3	11.3	624.2
Dry Storage Facility																					
Development																					0.0
Implementation																					58.5
Operations	1.2	1.2	1.8	1.8	1.8	1.8	1.8	1.8	1.8	1.8	1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.2			61.1
Decontam & Decomm																			1.2	1.2	2.5
Subtotal	1.2	1.2	1.8	1.8	1.8	1.8	1.8	1.8	1.8	1.8	1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.2	122.1
Hot Vacuum Drying Facility																					
Development																					0.0
Implementation																					31.2
Operations																					25.0
Decontam & Decomm																			2.5	2.5	5.0
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.5	2.5	61.2
Co-Disposal Canisters																					
Development																					0.0
Implementation																					15.0
Operations	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2		27.3
Decontam & Decomm																					0.0
Subtotal	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.0	42.3
Transportation System																					
Development																					0.0
Implementation		4.0																			10.0
Operations			3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	0.7	0.7	0.7	0.7	0.7	0.7	0.7	0.7			29.7
Decontam & Decomm																					0.0
Subtotal	0.0	4.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	0.7	0.7	0.7	0.7	0.7	0.7	0.7	0.7	0.0	0.0	39.7
Disposal (Repository)																					
Development (Fee)	5.0																				30.0
Implementation																					0.0
Operations			13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	3.3	3.3	3.3	3.3	3.3	3.3	3.3	3.3			130.0
Decontam & Decomm																					0.0
Subtotal	5.0	0.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	3.3	3.3	3.3	3.3	3.3	3.3	3.3	3.3	0.0	0.0	160.0
System Total																					
Development	5.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	45.0
Implementation	0.0	4.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	315.6
Operations	7.1	7.1	29.4	29.4	29.4	29.4	29.4	29.4	29.4	29.4	11.1	11.1	11.1	11.1	11.1	11.1	11.1	11.1	0.0	0.0	1005.5
Decontam & Decomm	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	15.0	15.0	30.0
Subtotal	12.1	11.1	29.4	29.4	29.4	29.4	29.4	29.4	29.4	29.4	11.1	11.1	11.1	11.1	11.1	11.1	11.1	11.1	15.0	15.0	1396.1

8.5 Melt and Dilute

In addition to the functions of the dry transfer facility, a melt and dilute module is conceptualized to perform the functions associated with this technology. The basis for the estimate of this module is WSRC-TR-95-0180 Life Cycle Cost Estimates for Disposal of Aluminum-Clad HEU SNF (U), scaled up from the value for the melt and dilute option in that study based on higher throughput. Table C8.5-1 shows the estimated cost for additional equipment for the Melt and Dilute module.

Table C8.5-1 Additional Equipment for Melt and Dilute Option

<u>Equipment</u>	<u>Cost (Thousands of 1996 dollars)</u>
Furnaces	1,318
Offgas System	1,070
Misc. Equipment	8,310
Allowances	4,513
Total Melt & Dilute	15,211

Additional building space for this treatment module is 2400 ft² of hot cell space, costed at \$2306 per square foot (1996 dollars), consistent with the transfer facility basis, for an additional building cost of \$5,534,000.

Additional operating and maintenance costs were estimated based on WSRC-TR-95-0180 Life Cycle Cost Estimates for Disposal of Aluminum-Clad HEU SNF (U), scaled to twice the value of the melt and dilute option in that study based on higher throughput. The additional annual costs for a 15-shift workweek are shown in Table C8.5-2. Costs for other work patterns were scaled from this value.

Table C8.5-2 Additional Operating and Maintenance Costs for Melt and Dilute Option

<u>Item</u>	<u>Cost (Thousands of 1996 dollars)</u>
Labor	2,408
Utilities	166
Materials	3,180
Maintenance Contracts	4,232

Incremental additional cost for decontamination and decommissioning of the Melt and Dilute module is estimated to be \$4 million.

These costs are added to those of the transfer facility for the direct disposal options. Table C8.5-3 presents a summary of the total project costs for the transfer facility with the treatment module included, including design, construction, operation and decommissioning, including appropriate funding overheads. Table C8.5-4 summarizes overall costs for this option, and Table C8.5-5 shows the distribution of these costs in time.

Table C8.5-3 Receipt and Treatment Facility - Melt & Dilute

Item	Description	Estimated Cost
DEVELOPMENT COSTS		
0.0	Development	
0.1	Research and Development	10,000
0.2	Process Demonstration	10,000
0.3	Waste Form Qualification	0
0.4	Development Contingency (0% of 0.1, 0.2, and 0.3)	
	SUBTOTAL 0.0 DEVELOPMENT	20,000

IMPLEMENTATION COSTS		
1.0	Construction Costs (Fixed Price Contract)	
1.1	Building Structure Costs	41,933
1.2	Building Equipment Costs	45,350
1.3	Construction Indirects (29% of 1.1 and 1.2)	25,312
	SUBTOTAL Construction 1.0	112,594
2.0	Title I and Title II Design (8% of 1.0)	9,008
3.0	Inspection (3% of 1.0)	3,378
4.0	Project Support (See Note 1) (6% of 1.0)	6,756
5.0	Construction Management (14% of 1.0)	15,763
6.0	SUBTOTAL Engr & Const (1.0, 2.0, 3.0, 4.0, 5.0)	147,498
7.0	Site G & A (16% of 6.0)	23,600
8.0	Total Estimated Cost (TEC) (6.0, 7.0)	171,098
9.0	OPC (Other Project Costs) (See Note 2) (30% of 8.0)	51,329
10.0	Implementation Contingency (0% of 8.0 and 9.0)	
11.0	Total Projected Cost (8.0, 9.0, 10.0) IMPLEMENTATION	222,428

OPERATIONS COSTS					
		5 Shifts / Week	10 Shifts / Week	15 Shifts / Week	21 Shifts / Week
12.0	Operations and Maintenance Costs (See Note 3)				
12.1	Annual Operation Costs (Labor)	1,263	2525	3788	5303
12.2	Annual Utility Costs	127	255	382	535
12.3	Annual Material / Consumable Costs	1,434	2868	4302	6022
12.4	Annual Maintenance Costs	3,562	7124	10686	14960
12.5	Interim Storage Costs				
12.6	Operations Contingency (0% of 12.1 through 12.5)				
	SUBTOTAL 12.0 OPERATIONS (12.1 through 12.6).	6,386	12,772	19,158	26,821

DECONTAMINATION AND DECOMMISSIONING COSTS		
13.0	Decontamination and Decommissioning Costs	
13.1	Facility and Equipment D&D	22,530
13.2	Storage D&D	
	SUBTOTAL 13.0 D&D	22,530

NOTES:

1. Project Support includes Project Management, Project Control, Estimating and QA
2. Other Project Costs include NEPA process, developing operating procedures, startup and checkout.
3. Annual costs may vary with years depending on operations performed.
4. Percentages shown above are suggested starting point values, and may be adjusted.

**Table C8.5-4
Total System Cost (\$million) Melt and Dilute (400 Co-Disposal Pkgs)**

	Cost Items	R&D/ Development	Facility & Equipment	Operations	D & D	Total
A	WET STORAGE AND HANDLING - Wet Transfer and Storage	0.0	0.0	346.6 346.6	0.0	346.6
B	TRANSFER AND PACKAGING - Treatment Facility (Recv & Pkg) - Canisters	0.0	184.6 169.6 15.0	189.4 181.0 8.4	18.5 18.5	392.5
C	TREATMENT - Treatment Facility (Treatment Module)	20.0 20.0	52.9 52.9	197.1 197.1	4.0 4.0	273.9
D	INTERIM STORAGE - Vault Storage	0.0	41.1 41.1	54.4 54.4	1.4 1.4	96.9
E	DISPOSAL E1 - Transportation E2 - Repository Fee E3 - Repository Operations	30.0 30.0	10.0 10.0	49.1 9.1 40.0	0.0	89.1
F	CREDITS	0.0	0.0	0.0	0.0	0.0
	TOTALS	50.0	288.6	836.6	24.0	1199.1

Table C8.5-5 Annual Cost Summary -- Time Phase Melt & Dilute

Year	1996	1997	1998	1999	2000	2001	2002	2003	2004	2005	2006	2007	2008	2009	2010	2011	2012	2013	2014	2015	2016
Wet Storage and Transfer																					
RBOF Operations	15.0	15.0	15.0	15.0	15.0	15.0	15.0	8.2	8.2	8.2	8.2	8.2	8.2								
L-Basin Operations	17.2	17.2	17.2	17.2	17.2	17.2	17.2	12.0	12.0	12.0	12.0	12.0	12.0								
Subtotal	32.2	32.2	32.2	32.2	32.2	32.2	32.2	20.2	20.2	20.2	20.2	20.2	20.2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Receipt and Treatment Facility																					
Development		3.0	3.0	6.0	6.0	2.0															
Implementation				7.8	60.2	90.3	64.0														
Operations								26.8	26.8	26.8	26.8	26.8	26.8	12.8	12.8	12.8	12.8	12.8	12.8	12.8	6.4
Decontam & Decomm																					
Subtotal	0.0	3.0	3.0	13.8	66.2	92.3	64.0	26.8	26.8	26.8	26.8	26.8	26.8	12.8	12.8	12.8	12.8	12.8	12.8	12.8	6.4
Dry Storage Facility																					
Development																					
Implementation				1.4	11.1	16.7	11.8														
Operations								2.9	2.9	2.9	2.9	2.9	2.9	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.0
Decontam & Decomm																					
Subtotal	0.0	0.0	0.0	1.4	11.1	16.7	11.8	2.9	2.9	2.9	2.9	2.9	2.9	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.0
Co-Disposal Canisters																					
Development																					
Implementation					5.0	5.0	5.0														
Operations								0.8	0.8	0.8	0.8	0.8	0.8	0.3	0.3	0.3	0.3	0.3	0.3	0.3	0.1
Decontam & Decomm																					
Subtotal	0.0	0.0	0.0	0.0	5.0	5.0	5.0	0.8	0.8	0.8	0.8	0.8	0.8	0.3	0.3	0.3	0.3	0.3	0.3	0.3	0.1
Transportation System																					
Development																					
Implementation					2.0	2.0	2.0														
Operations																					
Decontam & Decomm																					
Subtotal	0.0	0.0	0.0	0.0	2.0	2.0	2.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Disposal (Repository)																					
Development (Fee)																					
Implementation																				5.0	5.0
Operations																				5.0	5.0
Decontam & Decomm																				5.0	5.0
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.0	5.0
System Total																					
Development	0.0	3.0	3.0	6.0	6.0	2.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.0	5.0	5.0
Implementation	0.0	0.0	0.0	9.3	78.4	114.0	82.9	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Operations	32.2	32.2	32.2	32.2	32.2	32.2	32.2	50.8	50.8	50.8	50.8	50.8	50.8	14.7	14.7	14.7	14.7	14.7	14.7	14.7	7.5
Decontam & Decomm	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Subtotal	32.2	35.2	35.2	47.5	116.6	148.2	115.1	50.8	50.8	50.8	50.8	50.8	50.8	14.7	14.7	14.7	14.7	14.7	19.7	19.7	12.5

Table C8.5-5 Annual Cost Summary -- Time Phase Melt & Dilute

Year	2017	2018	2019	2020	2021	2022	2023	2024	2025	2026	2027	2028	2029	2030	2031	2032	2033	2034	2035	2036	2037	Total
Wet Storage and Transfer																						
RBOF Operations																						154.2
L-Basin Operations																						192.4
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	346.6
Receipt and Treatment Facility																						
Development																						20.0
Implementation																						222.4
Operations	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	378.0
Decontam & Decomm																				11.3	11.3	22.5
Subtotal	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	11.3	11.3	643.0
Dry Storage Facility																						
Development																						0.0
Implementation																						41.1
Operations	1.0	1.0	1.0	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	54.4
Decontam & Decomm																				0.7	0.7	1.4
Subtotal	1.0	1.0	1.0	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.0	1.0	1.0	1.0	1.0	1.0	1.0	0.7	0.7	96.9
Co-Disposal Canisters																						
Development																						0.0
Implementation																						15.0
Operations	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	8.4
Decontam & Decomm																						0.0
Subtotal	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.0	23.4
Transportation System																						
Development																						0.0
Implementation																						10.0
Operations			4.0	0.9	0.9	0.9	0.9	0.9	0.9	0.9	0.9	0.9	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	9.1
Decontam & Decomm																						0.0
Subtotal	0.0	0.0	4.0	0.9	0.9	0.9	0.9	0.9	0.9	0.9	0.9	0.9	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.0	19.1
Disposal (Repository)																						
Development (Fee)	5.0	5.0																				30.0
Implementation																						0.0
Operations				4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	40.0
Decontam & Decomm																						0.0
Subtotal	5.0	5.0	0.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	0.0	70.0
System Total																						
Development	5.0	5.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	50.0
Implementation	0.0	0.0	4.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	288.6
Operations	7.5	7.5	7.5	18.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	8.7	8.7	8.7	8.7	8.7	8.7	8.7	8.7	8.7	0.0	836.6
Decontam & Decomm	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	12.0	24.0
Subtotal	12.5	12.5	11.5	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	8.7	8.7	8.7	8.7	8.7	8.7	8.7	8.7	8.7	12.0	1199.1

8.6 Plasma Arc

In addition to the functions of the dry transfer facility, a plasma arc module is conceptualized to perform the functions associated with this technology. The plasma arc concept cost was developed based on a 1000 ft² hot cell for each unit operation (feed prep, two plasma arc furnaces, and canister decontamination).

Equipment lists were developed based on hot cell requirements from EGG-WM-10670 Waste Management Facilities Cost Information for Spent Nuclear Fuel (March 1993). Plasma arc furnace costs were provided by PNNL. Support equipment costs (canister decontamination) were based on DWPF experience. Table C8.6-1 shows the estimated cost for additional equipment for the Plasma Arc module.

Table C8.6-1 Additional Equipment for Plasma Arc Option

<u>Equipment</u>	<u>Cost (Thousands of 1996 dollars)</u>
Bulk Feed Prep	
Equipment	1,416
Allowances	758
Subtotal Bulk Feed Prep	2,174
Fuel Prep Cell	
Equipment	4,290
Allowances	1,943
Subtotal Fuel Prep	6,233
Plasma Arc Cell	
Plasma Arc Systems	20,203
Canister Handling	8,938
Misc. Equipment	20,745
Allowances	14,844
Subtotal Plasma Arc Cell	64,730
Total Plasma Arc Module	73,137

Additional building space for this plasma arc treatment module is 4000 ft² of hot cell space, costed at \$2306 per square foot (1996 dollars), and 4000 ft² of makeup and gallery space, costed at \$461 per square foot (1996 dollars), consistent with the transfer facility basis, for an additional building cost of \$11,068,000.

Additional operating and maintenance costs were estimated based on WSRC-TR-95-0180 Life Cycle Cost Estimates for Disposal of Aluminum-Clad HEU SNF (U), scaled to twice the value of the melt and dilute option in that study based on higher throughput. The additional annual costs for a 15-shift workweek are shown in Table C8.3-2. Costs for other work patterns were scaled from this value.

Table C8.6-2 Additional Operating and Maintenance Costs for Plasma Arc Option

<u>Item</u>	<u>Cost (Thousands of 1996 dollars)</u>
Labor	2,408
Utilities	166
Materials	3,180
Maintenance Contracts	4,232

Incremental additional cost for decontamination and decommissioning of the Plasma Arc module is estimated to be \$4 million.

These costs are added to those of the transfer facility for the direct disposal options. Table C8.6-3 presents a summary of the total project costs for the transfer facility with the treatment module included, including design, construction, operation and decommissioning, including appropriate funding overheads. Table C8.6-4 summarizes overall costs for this option, and Table C8.6-5 shows the distribution of these costs in time.

Table C8.6-3 Storage Facility Estimate - Plasma Arc

Item	Description	Estimated Cost			
DEVELOPMENT COSTS					
0.0	Development				
0.1	Research and Development	20,000			
0.2	Process Demonstration	30,000			
0.3	Waste Form Qualification	0			
0.4	Development Contingency (0% of 0.1, 0.2, and 0.3)				
	SUBTOTAL 0.0 DEVELOPMENT	50,000			
IMPLEMENTATION COSTS					
1.0	Construction Costs (Fixed Price Contract)				
1.1	Building Structure Costs	47,467			
1.2	Building Equipment Costs	103,275			
1.3	Construction Indirects (29% of 1.1 and 1.2)	43,715			
	SUBTOTAL Construction 1.0	194,457			
2.0	Title I and Title II Design (8% of 1.0)	15,557			
3.0	Inspection (3% of 1.0)	5,834			
4.0	Project Support (See Note 1) (6% of 1.0)	11,667			
5.0	Construction Management (14% of 1.0)	27,224			
6.0	SUBTOTAL Engr & Const (1.0, 2.0, 3.0, 4.0, 5.0)	254,738			
7.0	Site G & A (16% of 6.0)	40,758			
8.0	Total Estimated Cost (TEC) (6.0, 7.0)	295,496			
9.0	OPC (Other Project Costs) (See Note 2) (30% of 8.0)	88,649			
10.0	Implementation Contingency (0% of 8.0 and 9.0)				
11.0	Total Projected Cost (8.0, 9.0, 10.0) IMPLEMENTATION	384,145			
OPERATIONS COSTS					
12.0	Operations and Maintenance Costs (See Note 3)		5 Shifts / Week	10 Shifts / Week	15 Shifts / Week
12.1	Annual Operation Costs (Labor)	1,263		2,525	3,788
12.2	Annual Utility Costs	127		255	382
12.3	Annual Material / Consumable Costs	1,434		2,868	4,302
12.4	Annual Maintenance Costs	3,562		7,124	10,686
12.5	Interim Storage Costs				
12.6	Operations Contingency (0% of 12.1 through 12.5)				
	SUBTOTAL 12.0 OPERATIONS (12.1 through 12.6).	6,386		12,772	19,158
					26,821
DECONTAMINATION AND DECOMMISSIONING COSTS					
13.0	Decontamination and Decommissioning Costs				
13.1	Facility and Equipment D&D	22,530			
13.2	Storage D&D				
	SUBTOTAL 13.0 D&D	22,530			

NOTES:

1. Project Support includes Project Management, Project Control, Estimating and QA
2. Other Project Costs include NEPA process, developing operating procedures, startup and checkout.
3. Annual costs may vary with years depending on operations performed.
4. Percentages shown above are suggested starting point values, and may be adjusted.

**Table C8.6-4
Task Team Cost Model (\$million) Plasma Arc (400 Co-Disposal Pkgs)**

	Cost Items	R&D / Development	Facility & Equipment	Operations	D & D	Total
A	WET STORAGE AND HANDLING - Wet Transfer and Storage	0.0	0.0	456.4 456.4	0.0	456.4
B	TRANSFER AND PACKAGING - Treatment Facility (Recv and Pkg) - Canisters	0.0	184.6 169.6 15.0	177.3 169.4 8.0	18.5 18.5	380.4
C	TREATMENT - Treatment Facility (Treatment Module)	50.0 50.0	214.6 214.6	184.4 184.4	4.0 4.0	453.0
D	INTERIM STORAGE - Vault Storage	0.0	41.1 41.1	50.8 50.8	1.4 1.4	93.3
E	DISPOSAL E1 - Transportation E2 - Repository Fee E3 - Repository Operations	30.0 30.0	10.0 10.0	49.1 9.1 40.0	0.0	89.1
F	CREDITS	0.0	0.0	0.0	0.0	0.0
	TOTALS	80.0	450.3	918.1	24.0	1472.3

Table C8.6-5 Annual Cost Summary -- Time Phase Plasma Arc

Year	1996	1997	1998	1999	2000	2001	2002	2003	2004	2005	2006	2007	2008	2009	2010	2011	2012	2013	2014	2015	2016	2017
Wet Storage and Transfer																						
RBOF Operations	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	8.2	8.2	8.2	8.2	8.2	8.2	8.2	8.2					
L-Basin Operations	17.2	17.2	17.2	17.2	17.2	17.2	17.2	22.2	17.2	12.0	12.0	12.0	12.0	12.0	12.0	12.0	12.0					
Subtotal	32.2	32.2	32.2	32.2	32.2	32.2	32.2	37.2	32.2	20.2	20.2	20.2	20.2	20.2	20.2	20.2	20.2	0.0	0.0	0.0	0.0	0.0
Receipt and Treatment Facility																						
Development		5.0	5.0	5.0	10.0	15.0	5.0	5.0														
Implementation						17.2	20.3	78.4	157.7	110.6												
Operations											26.8	26.8	26.8	26.8	26.8	26.8	26.8	12.8	12.8	12.8	6.4	6.4
Decontam & Decomm																						
Subtotal	0.0	5.0	5.0	5.0	10.0	32.2	25.3	83.4	157.7	110.6	26.8	26.8	26.8	26.8	26.8	26.8	26.8	12.8	12.8	12.8	6.4	6.4
Dry Storage Facility																						
Development																						
Implementation						1.8	2.2	8.4	16.9	11.8												
Operations											2.9	2.9	2.9	2.9	2.9	2.9	2.9	1.6	1.6	1.6	1.0	1.0
Decontam & Decomm																						
Subtotal	0.0	0.0	0.0	0.0	0.0	1.8	2.2	8.4	16.9	11.8	2.9	2.9	2.9	2.9	2.9	2.9	2.9	1.6	1.6	1.6	1.0	1.0
Co-Disposal Canisters																						
Development																						
Implementation								5.0	5.0	5.0												
Operations											0.8	0.8	0.8	0.8	0.8	0.8	0.8	0.4	0.4	0.4	0.0	0.0
Decontam & Decomm																						
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.0	5.0	5.0	0.8	0.8	0.8	0.8	0.8	0.8	0.8	0.4	0.4	0.4	0.0	0.0
Transportation System																						
Development																						
Implementation								2.0	2.0	2.0												
Operations																						
Decontam & Decomm																						
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.0	2.0	2.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Disposal (Repository)																						
Development (Fee)																						
Implementation																						
Operations																						
Decontam & Decomm																						
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.0	5.0	5.0	5.0	5.0
System Total																						
Development	0.0	5.0	5.0	5.0	10.0	15.0	5.0	5.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.0	5.0	5.0	5.0	5.0
Implementation	0.0	0.0	0.0	0.0	0.0	19.0	22.5	93.8	181.6	129.4	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Operations	32.2	32.2	32.2	32.2	32.2	32.2	32.2	37.2	32.2	20.2	50.8	50.8	50.8	50.8	50.8	50.8	50.8	14.8	14.8	14.8	7.5	7.5
Decontam & Decomm	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Subtotal	32.2	37.2	37.2	37.2	42.2	66.2	59.7	136.0	213.8	149.6	50.8	50.8	50.8	50.8	50.8	50.8	50.8	19.8	19.8	19.8	12.5	12.5

Table C8.6-5 Annual Cost Summary -- Time Phase Plasma Arc

2018	2019	2020	2021	2022	2023	2024	2025	2026	2027	2028	2029	2030	2031	2032	2033	2034	2035	2036	2037	Total
																				200.6
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	255.8
																				50.0
6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4			384.1
6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	11.3	11.3	353.8
																				22.5
6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	11.3	11.3	810.5
																				0.0
1.0	1.0	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.0	1.0	1.0	1.0	1.0	1.0	1.0			41.1
1.0	1.0	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.0	1.0	1.0	1.0	1.0	1.0	1.0	0.7	0.7	50.8
																				1.4
1.0	1.0	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.0	1.0	1.0	1.0	1.0	1.0	1.0	0.7	0.7	93.3
																				0.0
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0			15.0
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0			8.0
																				0.0
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	23.0
																				0.0
	4.0																			10.0
		0.9	0.9	0.9	0.9	0.9	0.9	0.9	0.9	0.9	0.2	0.2	0.2	0.2	0.2	0.2	0.2			9.1
																				0.0
0.0	4.0	0.9	0.9	0.9	0.9	0.9	0.9	0.9	0.9	0.9	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.0	0.0	19.1
																				0.0
5.0																				30.0
			4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0			0.0
																				40.0
																				0.0
5.0	0.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	0.0	0.0	70.0
																				0.0
5.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	80.0
0.0	4.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	450.3
7.5	7.5	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	8.7	8.7	8.7	8.7	8.7	8.7	8.7	0.0	0.0	918.1
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	12.0	24.0
12.5	11.5	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	8.7	8.7	8.7	8.7	8.7	8.7	8.7	12.0	12.0	1472.3

8.7 GMODS

In addition to the functions of the dry transfer facility, a GMODS module is conceptualized to perform the functions associated with this technology. The GMODS concept cost was developed based on a 1000 ft² hot cell for each unit operation (feed prep, two melters, and canister decontamination).

Equipment lists were developed based on hot cell requirements from EGG-WM-10670 Waste Management Facilities Cost Information for Spent Nuclear Fuel (March 1993). Specialized GMODS equipment costs were provided by ORNL. Support equipment costs (canister decontamination) were based on DWPF experience. Table C8.7-1 shows the estimated cost for additional equipment for the GMODS module.

Table C8.7-1 Additional Equipment for GMODS Option

<u>Equipment</u>	<u>Cost (Thousands of 1996 dollars)</u>
Bulk Feed Prep	
Equipment	1,889
Allowances	944
Subtotal Bulk Feed Prep	2,833
Fuel Prep Cell	
Equipment	3,923
Allowances	2,310
Subtotal Fuel Prep	6,233
Melter Cell	
Melters	10,376
Canister Handling	5,600
Misc. Equipment	18,172
Allowances	12,034
Subtotal Melter Cell	46,182
Total GMODS Module	55,248

Additional building space for this GMODS treatment module is 4000 ft² of hot cell space, costed at \$2306 per square foot (1996 dollars), and 4,000 ft² of makeup and gallery space, costed at \$461 per square foot (1996 dollars), consistent with the transfer facility basis, for an additional building cost of \$11,068,000.

Additional operating and maintenance costs were estimated based on WSRC-TR-95-0180 Life Cycle Cost Estimates for Disposal of Aluminum-Clad HEU SNF (U), scaled to twice the value of the dissolve, dilute and vitrify option in that study based on higher throughput. The additional annual costs for a 15-shift workweek are shown in Table C8.7-2. Costs for other work patterns were scaled from this value.

Table C8.7-2 Additional Operating and Maintenance Costs for GMODS Option

<u>Item</u>	<u>Cost (Thousands of 1996 dollars)</u>
Labor	2,408
Utilities	166
Materials	3,180
Maintenance Contracts	4,232

Incremental additional cost for decontamination and decommissioning of the GMODS module is estimated to be \$4 million.

These costs are added to those of the transfer facility for the direct disposal options. Table C8.7-3 presents a summary of the total project costs for the transfer facility with the treatment module included, including design, construction, operation and decommissioning, including appropriate funding overheads. Table C8.7-4 summarizes overall costs for this option, and Table C8.7-5 shows the distribution of these costs in time.

Table C8.7-3 Storage Facility Estimate - GMODs

Item	Description	Estimated Cost				
DEVELOPMENT COSTS						
0.0	Development					
0.1	Research and Development	25,000				
0.2	Process Demonstration	25,000				
0.3	Waste Form Qualification	0				
0.4	Development Contingency (0% of 0.1, 0.2, and 0.3)					
	SUBTOTAL 0.0 DEVELOPMENT	50,000				
IMPLEMENTATION COSTS						
1.0	Construction Costs (Fixed Price Contract)					
1.1	Building Structure Costs	47,467				
1.2	Building Equipment Costs	85,386				
1.3	Construction Indirects (29% of 1.1 and 1.2)	38,527				
	SUBTOTAL Construction 1.0	171,380				
2.0	Title I and Title II Design (8% of 1.0)	13,710				
3.0	Inspection (3% of 1.0)	5,141				
4.0	Project Support (See Note 1) (6% of 1.0)	10,283				
5.0	Construction Management (14% of 1.0)	23,993				
6.0	SUBTOTAL Engr & Const (1.0, 2.0, 3.0, 4.0, 5.0)	224,508				
7.0	Site G & A (16% of 6.0)	35,921				
8.0	Total Estimated Cost (TEC) (6.0, 7.0)	260,429				
9.0	OPC (Other Project Costs) (See Note 2) (30% of 8.0)	78,129				
10.0	Implementation Contingency (0% of 8.0 and 9.0)					
11.0	Total Projected Cost (8.0, 9.0, 10.0) IMPLEMENTATION	338,557				
OPERATIONS COSTS						
12.0	Operations and Maintenance Costs (See Note 3)		5 Shifts / Week	10 Shifts / Week	15 Shifts / Week	21 Shifts / Week
12.1	Annual Operation Costs (Labor)	1,263	2,525	3,788	5,303	
12.2	Annual Utility Costs	127	255	382	535	
12.3	Annual Material / Consumable Costs	1,434	2,868	4,302	6,022	
12.4	Annual Maintenance Costs	3,562	7,124	10,686	14,960	
12.5	Interim Storage Costs					
12.6	Operations Contingency (0% of 12.1 through 12.5)					
	SUBTOTAL 12.0 OPERATIONS (12.1 through 12.6)	6,386	12,772	19,158	26,821	
DECONTAMINATION AND DECOMMISSIONING COSTS						
13.0	Decontamination and Decommissioning Costs					
13.1	Facility and Equipment D&D	22,530				
13.2	Storage D&D					
	SUBTOTAL 13.0 D&D	22,530				

NOTES:

1. Project Support includes Project Management, Project Control, Estimating and QA
2. Other Project Costs include NEPA process, developing operating procedures, startup and checkout.
3. Annual costs may vary with years depending on operations performed.
4. Percentages shown above are suggested starting point values, and may be adjusted.

**Table C8.7-4
Total System Cost (\$million) GMODS (800 Co-Disposal Pkgs)**

	Cost Items	R&D/ Development	Facility & Equipment	Operations	D & D	Total
A	WET STORAGE AND HANDLING - Wet Transfer and Storage	0.0	0.0	456.4 456.4	0.0	456.4
B	TRANSFER AND PACKAGING - Treatment Facility (Recv and Pkg) - Canisters	0.0	184.6 169.6 15.0	185.3 169.4 16.0	18.5 18.5	388.4
C	TREATMENT - Treatment Facility (Treatment Module)	50.0 50.0	169.0 169.0	184.4 184.4	4.0 4.0	407.4
D	INTERIM STORAGE - Vault Storage	0.0	49.9 49.9	53.9 53.9	2.0 2.0	105.7
E	DISPOSAL E1 - Transportation E2 - Repository Fee E3 - Repository Operations	30.0 30.0	10.0 10.0	98.3 18.3 80.0	0.0	138.3
F	CREDITS	0.0	0.0	0.0	0.0	0.0
	TOTALS	80.0	413.4	978.3	24.5	1496.2

Table C8.7-5 Annual Cost Summary -- Time Phase GMODS

Year	1996	1997	1998	1999	2000	2001	2002	2003	2004	2005	2006	2007	2008	2009	2010	2011	2012	2013	2014	2015	2016	2017	
Wet Storage and Transfer																							
RBOF Operations	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	8.2	8.2	8.2	8.2	8.2	8.2	8.2	8.2						
L-Basin Operations	17.2	17.2	17.2	17.2	17.2	17.2	17.2	22.2	17.2	12.0	12.0	12.0	12.0	12.0	12.0	12.0	12.0						
Subtotal	32.2	32.2	32.2	32.2	32.2	32.2	32.2	37.2	32.2	20.2	20.2	20.2	20.2	20.2	20.2	20.2	20.2	0.0	0.0	0.0	0.0	0.0	
Receipt and Treatment Facility																							
Development		5.0	5.0	5.0	10.0	15.0	5.0	5.0															
Implementation						15.1	17.9	69.1	139.0	97.5													
Operations											26.8	26.8	26.8	26.8	26.8	26.8	26.8	12.8	12.8	12.8	6.4	6.4	
Decontam & Decomm																							
Subtotal	0.0	5.0	5.0	5.0	10.0	30.1	22.9	74.1	139.0	97.5	26.8	26.8	26.8	26.8	26.8	26.8	26.8	12.8	12.8	12.8	6.4	6.4	
Dry Storage Facility																							
Development																							
Implementation						2.2	2.6	10.2	20.5	14.4													
Operations											3.1	3.1	3.1	3.1	3.1	3.1	3.1	1.7	1.7	1.7	1.1	1.1	
Decontam & Decomm																							
Subtotal	0.0	0.0	0.0	0.0	0.0	2.2	2.6	10.2	20.5	14.4	3.1	3.1	3.1	3.1	3.1	3.1	3.1	1.7	1.7	1.7	1.1	1.1	
Co-Disposal Canisters																							
Development																							
Implementation								5.0	5.0	5.0													
Operations											1.7	1.7	1.7	1.7	1.7	1.7	1.7	0.8	0.8	0.8	0.1	0.1	
Decontam & Decomm																							
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.0	5.0	5.0	1.7	1.7	1.7	1.7	1.7	1.7	1.7	0.8	0.8	0.8	0.1	0.1	
Transportation System																							
Development																							
Implementation								2.0	2.0	2.0													
Operations																							
Decontam & Decomm																							
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.0	2.0	2.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	
Disposal (Repository)																							
Development (Fee)																			5.0	5.0	5.0	5.0	5.0
Implementation																							
Operations																							
Decontam & Decomm																							
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.0	5.0	5.0	5.0	
System Total																							
Development	0.0	5.0	5.0	5.0	10.0	15.0	5.0	5.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.0	5.0	5.0	5.0	
Implementation	0.0	0.0	0.0	0.0	0.0	17.4	20.5	86.3	166.5	118.8	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	
Operations	32.2	32.2	32.2	32.2	32.2	32.2	32.2	37.2	32.2	20.2	51.8	51.8	51.8	51.8	51.8	51.8	51.8	15.3	15.3	15.3	7.6	7.6	
Decontam & Decomm	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	
Subtotal	32.2	37.2	37.2	37.2	42.2	64.6	57.7	128.5	198.7	139.0	51.8	51.8	51.8	51.8	51.8	51.8	51.8	20.3	20.3	20.3	12.6	12.6	

Table C8.7-5 Annual Cost Summary -- Time Phase GMODS

Year	2018	2019	2020	2021	2022	2023	2024	2025	2026	2027	2028	2029	2030	2031	2032	2033	2034	2035	2036	2037	Total
Wet Storage and Transfer																					
RBOF Operations																					200.6
L-Basin Operations																					255.8
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	456.4
Receipt and Treatment Facility																					
Development																					50.0
Implementation																					338.6
Operations	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4			353.8
Decontam & Decomm																			11.3	11.3	22.5
Subtotal	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	6.4	11.3	11.3	764.9
Dry Storage Facility																					
Development																					0.0
Implementation																					49.9
Operations	1.1	1.1	1.7	1.7	1.7	1.7	1.7	1.7	1.7	1.7	1.1	1.1	1.1	1.1	1.1	1.1	1.1	1.1			53.9
Decontam & Decomm																			1.0	1.0	2.0
Subtotal	1.1	1.1	1.7	1.7	1.7	1.7	1.7	1.7	1.7	1.7	1.1	1.1	1.1	1.1	1.1	1.1	1.1	1.1	1.0	1.0	105.7
Co-Disposal Canisters																					
Development																					0.0
Implementation																					15.0
Operations	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1			16.0
Decontam & Decomm																					0.0
Subtotal	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.0	0.0	31.0
Transportation System																					
Development																					0.0
Implementation		4.0																			10.0
Operations			1.8	1.8	1.8	1.8	1.8	1.8	1.8	1.8	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5			18.3
Decontam & Decomm																					0.0
Subtotal	0.0	4.0	1.8	1.8	1.8	1.8	1.8	1.8	1.8	1.8	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.0	0.0	28.3
Disposal (Repository)																					
Development (Fee)	5.0																				30.0
Implementation																					0.0
Operations			8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0			80.0
Decontam & Decomm																					0.0
Subtotal	5.0	0.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	0.0	0.0	110.0
System Total																					
Development	5.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	80.0
Implementation	0.0	4.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	413.4
Operations	7.6	7.6	18.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	0.0	0.0	978.3
Decontam & Decomm	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	12.2	12.2	24.5
Subtotal	12.6	11.6	18.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	12.2	12.2	1496.2

8.8 Dissolve and Vitrify

In addition to the functions of the dry transfer facility, a dissolve and vitrify module is conceptualized to perform the functions associated with this technology. The dissolve, dilute and vitrify concept cost was developed based on a 5400 ft² hot cell for the dissolution process, and a 2400 ft² hot cell for the melter. In addition, 7000 ft² of process space was added for the gallery and chemical makeup area (3000 sq. ft. and 4000 sq. ft. respectively). Equipment lists were developed based on hot cell requirements from EGG-WM-10670 Waste Management Facilities Cost Information for Spent Nuclear Fuel (March 1993). A DWPF melter and associated equipment was added to the equipment estimate for the dissolve and dilute option in WSRC-TR-95-0180. Support equipment costs (canister decontamination) were based on DWPF experience. Table C8.8-1 shows the estimated cost for additional equipment for the GMODS module.

Table C8.8-1 Additional Equipment for Dissolve and Vitrify Option

Equipment	Cost (Thousands of 1996 dollars)
Fuel Prep Cell	
Equipment	3,923
Allowances	2,310
Subtotal Fuel Prep	6,233
Dissolution Cell	
Tankage	3,140
Crane	1,515
Misc. Equipment	3,454
Allowances	12,002
Subtotal Dissolution Cell	20,112
Makeup Area	
Equipment	710
Allowances	237
Subtotal Makeup Area	948
Vitrification Cell	
Melter	31,568
Canister Handling	5,600
Misc. Equipment	12,573
Allowances	21,785
Subtotal Vitrification Cell	71,525
Total Dissolve and Vitrify Module	98,818

Additional building space for this Dissolve and Vitrify treatment module is a total of 7800 ft² of hot cell space, costed at \$2306 per square foot (1996 dollars), and 7,000 ft² of makeup and gallery space, costed at \$461 per square foot (1996 dollars), consistent with the transfer facility basis, for an additional building cost of \$21,213,000.

Additional operating and maintenance costs were estimated based on WSRC-TR-95-0180 Life Cycle Cost Estimates for Disposal of Aluminum-Clad HEU SNF (U), scaled to twice the value of the dissolve, dilute and vitrify option in that study based on higher throughput. The additional annual costs for a 15-shift workweek are shown in Table C8.8-2. Costs for other work patterns were scaled from this value.

Table C8.8-2 Additional Operating and Maintenance Costs for Dissolve and Vitrify Option

<u>Item</u>	<u>Cost (Thousands of 1996 dollars)</u>
Labor	7,280
Utilities	360
Materials	258
Maintenance Contracts	13,094

Incremental additional cost for decontamination and decommissioning of the GMODS module is estimated to be \$4 million.

These costs are added to those of the transfer facility for the direct disposal options. Table C8.8-3 presents a summary of the total project costs for the transfer facility with the treatment module included, including design, construction, operation and decommissioning, including appropriate funding overheads. Table C8.8-4 summarizes overall costs for this option, and Table C8.8-5 shows the distribution of these costs in time.

Table C8.8-3 Receipt and Treatment Facility - Dissolve and Vitrify

Item	Description	Estimated Cost			
DEVELOPMENT COSTS					
0.0	Development				
0.1	Research and Development	10,000			
0.2	Process Demonstration	10,000			
0.3	Waste Form Qualification	0			
0.4	Development Contingency (0% of 0.1, 0.2, and 0.3)				
	SUBTOTAL 0.0 DEVELOPMENT	20,000			
IMPLEMENTATION COSTS					
1.0	Construction Costs (Fixed Price Contract)				
1.1	Building Structure Costs	57,612			
1.2	Building Equipment Costs	128,956			
1.3	Construction Indirects (29% of 1.1 and 1.2)	54,105			
	SUBTOTAL Construction 1.0	240,673			
2.0	Title I and Title II Design (8% of 1.0)	19,254			
3.0	Inspection (3% of 1.0)	7,220			
4.0	Project Support (See Note 1) (6% of 1.0)	14,440			
5.0	Construction Management (14% of 1.0)	33,694			
6.0	SUBTOTAL Engr & Const (1.0, 2.0, 3.0, 4.0, 5.0)	315,281			
7.0	Site G & A (16% of 6.0)	50,445			
8.0	Total Estimated Cost (TEC) (6.0, 7.0)	365,727			
9.0	OPC (Other Project Costs) (See Note 2) (30% of 8.0)	109,718			
10.0	Implementation Contingency (0% of 8.0 and 9.0)				
11.0	Total Projected Cost (8.0, 9.0, 10.0) IMPLEMENTATION	475,444			
OPERATIONS COSTS					
12.0	Operations and Maintenance Costs (See Note 3)		5 Shifts / Week	10 Shifts / Week	15 Shifts / Week
12.1	Annual Operation Costs (Labor)	2,887	5,773	8,660	12,124
12.2	Annual Utility Costs	192	384	576	807
12.3	Annual Material / Consumable Costs	460	920	1,380	1,932
12.4	Annual Maintenance Costs	6,516	13,032	19,548	27,367
12.5	Interim Storage Costs				
12.6	Operations Contingency (0% of 12.1 through 12.5)				
	SUBTOTAL 12.0 OPERATIONS (12.1 through 12.6)	10,055	20,109	30,164	42,229
DECONTAMINATION AND DECOMMISSIONING COSTS					
13.0	Decontamination and Decommissioning Costs				
13.1	Facility and Equipment D&D	22,530			
13.2	Storage D&D				
	SUBTOTAL 13.0 D&D	22,530			

NOTES:

1. Project Support includes Project Management, Project Control, Estimating and QA
2. Other Project Costs include NEPA process, developing operating procedures, startup and checkout.
3. Annual costs may vary with years depending on operations performed.
4. Percentages shown above are suggested starting point values, and may be adjusted.

**Table C8.8-4
Total System Cost (\$million) Dissolve and Vitrify (800 Co-Disposal Pkgs)**

	Cost Items	R&D/ Development	Facility & Equipment	Operations	D & D	Total
A	WET STORAGE AND HANDLING - Wet Transfer and Storage	0.0	0.0	456.4 456.4	0.0	456.4
B	TRANSFER AND PACKAGING - Treatment Facility (Recv and Pkg) - Canisters	0.0	184.6 169.6 15.0	185.3 169.4 16.0	18.5 18.5	388.4
C	TREATMENT - Treatment Facility (Treatment Module)	20.0 20.0	305.9 305.9	387.7 387.7	4.0 4.0	717.5
D	INTERIM STORAGE - Vault Storage	0.0	49.9 49.9	53.9 53.9	2.0 2.0	105.7
E	DISPOSAL E1 - Transportation E2 - Repository Fee E3 - Repository Operations	30.0 30.0	10.0 10.0	98.3 18.3 80.0	0.0	138.3
F	CREDITS	0.0	0.0	0.0	0.0	0.0
	TOTALS	50.0	550.3	1181.6	24.5	1806.4

Table C8.8-5 Annual Cost Summary -- Time Phase D&V

Year	1996	1997	1998	1999	2000	2001	2002	2003	2004	2005	2006	2007	2008	2009	2010	2011	2012	2013	2014	2015	2016	2017
Wet Storage and Transfer																						
RBOF Operations	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	8.2	8.2	8.2	8.2	8.2	8.2	8.2	8.2					
L-Basin Operations	17.2	17.2	17.2	17.2	17.2	17.2	17.2	22.2	17.2	12.0	12.0	12.0	12.0	12.0	12.0	12.0	12.0					
Subtotal	32.2	32.2	32.2	32.2	32.2	32.2	32.2	37.2	32.2	20.2	20.2	20.2	20.2	20.2	20.2	20.2	20.2	0.0	0.0	0.0	0.0	0.0
Receipt and Treatment Facility																						
Development		2.0	2.0	2.0	4.0	6.0	2.0	2.0														
Implementation						21.3	25.1	97.0	195.2	136.9												
Operations											42.2	42.2	42.2	42.2	42.2	42.2	42.2	20.1	20.1	20.1	10.1	10.1
Decontam & Decomm																						
Subtotal	0.0	2.0	2.0	2.0	4.0	27.3	27.1	99.0	195.2	136.9	42.2	42.2	42.2	42.2	42.2	42.2	42.2	20.1	20.1	20.1	10.1	10.1
Dry Storage Facility																						
Development						2.2	2.6	10.2	20.5	14.4												
Implementation											3.1	3.1	3.1	3.1	3.1	3.1	3.1	1.7	1.7	1.7	1.1	1.1
Operations																						
Decontam & Decomm																						
Subtotal	0.0	0.0	0.0	0.0	0.0	2.2	2.6	10.2	20.5	14.4	3.1	3.1	3.1	3.1	3.1	3.1	3.1	1.7	1.7	1.7	1.1	1.1
Co-Disposal Canisters																						
Development																						
Implementation								5.0	5.0	5.0												
Operations											1.7	1.7	1.7	1.7	1.7	1.7	1.7	0.8	0.8	0.8	0.1	0.1
Decontam & Decomm																						
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.0	5.0	5.0	1.7	1.7	1.7	1.7	1.7	1.7	1.7	0.8	0.8	0.8	0.1	0.1
Adjustment for Relative Uncertainty																						
Development																						
Implementation								0.0	0.0	0.0												
Operations											0.0	0.0	0.0	0.0	0.0	0.0	0.0					
Decontam & Decomm																						
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Transportation System																						
Development																						
Implementation								2.0	2.0	2.0												
Operations																						
Decontam & Decomm																						
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.0	2.0	2.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Disposal (Repository)																						
Development (Fee)																						
Implementation																						
Operations																						
Decontam & Decomm																						
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.0	5.0	5.0	5.0	5.0
System Total																						
Development	0.0	2.0	2.0	2.0	4.0	6.0	2.0	2.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.0	5.0	5.0	5.0	5.0
Implementation	0.0	0.0	0.0	0.0	0.0	23.5	27.8	114.2	222.6	158.2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Operations	32.2	32.2	32.2	32.2	32.2	32.2	32.2	37.2	32.2	20.2	67.2	67.2	67.2	67.2	67.2	67.2	67.2	22.7	22.7	22.7	11.2	11.2
Decontam & Decomm	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Subtotal	32.2	34.2	34.2	34.2	36.2	61.7	62.0	153.4	254.8	178.4	67.2	67.2	67.2	67.2	67.2	67.2	67.2	27.7	27.7	27.7	16.2	16.2

Table C8.8-5 Annual Cost Summary -- Time Phase D&V

Year	2018	2019	2020	2021	2022	2023	2024	2025	2026	2027	2028	2029	2030	2031	2032	2033	2034	2035	2036	2037	Total	
Wet Storage and Transfer																						
RBOF Operations																					200.6	
L-Basin Operations																						255.8
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	456.4	
Receipt and Treatment Facility																						
Development																					20.0	
Implementation																					475.4	
Operations	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1			557.0	
Decontam & Decomm																			11.3	11.3	22.5	
Subtotal	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	11.3	11.3	1075.0	
Dry Storage Facility																						
Development																					0.0	
Implementation																					49.9	
Operations	1.1	1.1	1.7	1.7	1.7	1.7	1.7	1.7	1.7	1.7	1.1	1.1	1.1	1.1	1.1	1.1	1.1	1.1			53.9	
Decontam & Decomm																			1.0	1.0	2.0	
Subtotal	1.1	1.1	1.7	1.7	1.7	1.7	1.7	1.7	1.7	1.7	1.1	1.1	1.1	1.1	1.1	1.1	1.1	1.1	1.0	1.0	105.7	
Co-Disposal Canisters																						
Development																					0.0	
Implementation																					15.0	
Operations	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1			16.0	
Decontam & Decomm																					0.0	
Subtotal	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.0	0.0	31.0	
Adjustment for Relative Uncertainty																						
Development																					0.0	
Implementation																					0.0	
Operations																					0.0	
Decontam & Decomm																					0.0	
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	
Transportation System																						
Development																					0.0	
Implementation		4.0																			10.0	
Operations			1.8	1.8	1.8	1.8	1.8	1.8	1.8	1.8	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5			18.3	
Decontam & Decomm																					0.0	
Subtotal	0.0	4.0	1.8	1.8	1.8	1.8	1.8	1.8	1.8	1.8	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.0	0.0	28.3	
Disposal (Repository)																						
Development (Fee)	5.0																				30.0	
Implementation																					0.0	
Operations			8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0			80.0	
Decontam & Decomm																					0.0	
Subtotal	5.0	0.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	0.0	0.0	110.0	
System Total																						
Development	5.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	50.0	
Implementation	0.0	4.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	550.3	
Operations	11.2	11.2	21.7	21.7	21.7	21.7	21.7	21.7	21.7	21.7	13.7	13.7	13.7	13.7	13.7	13.7	13.7	13.7	0.0	0.0	1181.6	
Decontam & Decomm	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	12.2	12.2	24.5	
Subtotal	16.2	15.2	21.7	21.7	21.7	21.7	21.7	21.7	21.7	21.7	13.7	13.7	13.7	13.7	13.7	13.7	13.7	13.7	12.2	12.2	1806.4	

8.9 Electrometallurgical

In addition to the functions of the dry transfer facility, a Electrometallurgical treatment module is conceptualized to perform the functions associated with this technology. The electrometallurgy estimate was based on an air locked cell with an inert atmosphere for the EM process. Hot cell area of 800 sq. ft. was used for each unit process, or unit process group. An hot cell area of 400 sq. ft. was used for the two airlock. Total hot cell area was 6000 sq. ft. In addition 5,000 sq. ft. of gallery and chemical makeup area was provided.

Equipment lists were developed based on hot cell requirements from EGG-WM-10670 Waste Management Facilities Cost Information for Spent Nuclear Fuel (March 1993). Equipment estimates were also supported by ANL inputs. Support equipment costs (canister decontamination) were based on DWPF experience. Table C8.9-1 shows the estimated cost for additional equipment for the Electrometallurgical treatment module.

Table C8.9-1 Additional Equipment for Electrometallurgical Option

<u>Equipment</u>	<u>Cost (Thousands of 1996 dollars)</u>
Bulk Feed Prep	
Equipment	944
Allowances	545
Subtotal Bulk Feed Prep	1,489
Fuel Prep Cell	
Equipment	4,614
Allowances	2,743
Subtotal Fuel Prep	7,357
Airlock (2 each)	
Equipment	1,719
Allowances	773
Subtotal Airlock	2,492
Anode Prep Cell	
Furnaces	1,370
Misc. Equipment	3,699
Allowances Anode Prep	2,038
Subtotal Anode Prep	7,107

Electrorefiner Cell	
Electrorefiners	5,303
Misc. Equipment	618
Allowances	2,771
Subtotal Electrorefiner	8,693
U/Al Recovery/Package	
Furnaces	3,511
Misc. Equipment	1,035
Allowances	1,879
Subtotal U/Al Recovery	6,425
Oxidation Cell	
Furnaces	628
Misc. Equipment	1,042
Allowances	702
Subtotal Oxidation	2,372
Glass Melter Cell	
Glass Melter/Marble Machine	799
Canister Handling	442
Misc. Equipment	3,874
Allowances	1,850
Subtotal Glass Melter	6,966
 Total Electrometallurgy	 42,898

Additional building space for this Dissolve and Vitrify treatment module is a total of 6,000 ft² of hot cell space, costed at \$2306 per square foot (1996 dollars), and 5,000 ft² of makeup and gallery space, costed at \$461 per square foot (1996 dollars), consistent with the transfer facility basis, for an additional building cost of \$16,141,000.

Because of the number of unit operations, Electrometallurgical processing was considered to be most similar to dissolve, dilute and vitrify. Additional operating and maintenance costs were estimated based on WSRC-TR-95-0180 Life Cycle Cost Estimates for Disposal of Aluminum-Clad HEU SNF (U), scaled to twice the value of the dissolve, dilute and vitrify option in that study based on higher throughput. The additional annual costs for a 15-shift workweek are shown in Table C8.9-2. Costs for other work patterns were scaled from this value.

Table C8.9-2 Additional Operating and Maintenance Costs for Dissolve and Vitrify Option

Item	Cost (Thousands of 1996 dollars)
Labor	7,280
Utilities	360
Materials	258
Maintenance Contracts	13,094

Incremental additional cost for decontamination and decommissioning of the GMODS module is estimated to be \$4 million.

These costs are added to those of the transfer facility for the direct disposal options. Table C8.9-3 presents a summary of the total project costs for the transfer facility with the treatment module included, including design, construction, operation and decommissioning, including appropriate funding overheads. Table C8.9-4 summarizes overall costs for this option, and Table C8.9-5 shows the distribution of these costs in time.

Table C8.9-3 Storage Facility Estimate - EM

Item	Description	Estimated Cost			
DEVELOPMENT COSTS					
0.0	Development				
0.1	Research and Development	20,000			
0.2	Process Demonstration	20,000			
0.3	Waste Form Qualification	0			
0.4	Development Contingency (0% of 0.1, 0.2, and 0.3)				
	SUBTOTAL 0.0 DEVELOPMENT	40,000			
IMPLEMENTATION COSTS					
1.0	Construction Costs (Fixed Price Contract)				
1.1	Building Structure Costs	52,539			
1.2	Building Equipment Costs	73,036			
1.3	Construction Indirects (29% of 1.1 and 1.2)	36,417			
	SUBTOTAL Construction 1.0	161,992			
2.0	Title I and Title II Design (8% of 1.0)	12,959			
3.0	Inspection (3% of 1.0)	4,860			
4.0	Project Support (See Note 1) (6% of 1.0)	9,720			
5.0	Construction Management (14% of 1.0)	22,679			
6.0	SUBTOTAL Engr & Const (1.0, 2.0, 3.0, 4.0, 5.0)	212,210			
7.0	Site G & A (16% of 6.0)	33,954			
8.0	Total Estimated Cost (TEC) (6.0, 7.0)	246,163			
9.0	OPC (Other Project Costs) (See Note 2) (30% of 8.0)	73,849			
10.0	Implementation Contingency (0% of 8.0 and 9.0)				
11.0	Total Projected Cost (8.0, 9.0, 10.0) IMPLEMENTATION	320,012			
OPERATIONS COSTS					
12.0	Operations and Maintenance Costs (See Note 3)		5 Shifts / Week	10 Shifts / Week	15 Shifts / Week
12.1	Annual Operation Costs (Labor)	2,887		5773	8660
12.2	Annual Utility Costs	192		384	576
12.3	Annual Material / Consumable Costs	460		920	1380
12.4	Annual Maintenance Costs	6,516		13032	19548
12.5	Interim Storage Costs				
12.6	Operations Contingency (0% of 12.1 through 12.5)				
	SUBTOTAL 12.0 OPERATIONS (12.1 through 12.6).	10,055	20,109	30,164	42,229
DECONTAMINATION AND DECOMMISSIONING COSTS					
13.0	Decontamination and Decommissioning Costs				
13.1	Facility and Equipment D&D	22,530			
13.2	Storage D&D				
	SUBTOTAL 13.0 D&D	22,530			

NOTES:

1. Project Support includes Project Management, Project Control, Estimating and QA
2. Other Project Costs include NEPA process, developing operating procedures, startup and checkout.
3. Annual costs may vary with years depending on operations performed.
4. Percentages shown above are suggested starting point values, and may be adjusted.

**Table C8.9-4
Total System Cost (\$million) Electrometallurgical (90 Glass Logs)**

	Cost Items	R&D/ Development	Facility & Equipment	Operations	D & D	Total
A	WET STORAGE AND HANDLING - Wet Transfer and Storage	0.0	0.0	436.2 436.2	0.0	436.2
B	TRANSFER AND PACKAGING - Treatment Facility (Recv and Pkg) - Canisters	0.0	169.6 169.6	176.4 175.5 0.9	18.5 18.5	364.5
C	TREATMENT - Treatment Facility (Treatment Module)	40.0 40.0	150.5 150.5	401.6 401.6	4.0 4.0	596.1
D	INTERIM STORAGE - Vault Storage - GWSB Incremental Charges	0.0	3.0 0.0 3.0	0.0 0.0	0.0 0.0	3.0
E	DISPOSAL E1 - Transportation E2 - Repository Fee E3 - Repository Operations	30.0	10.0 10.0	14.4 3.6 10.8	0.0	54.4
F	CREDITS - Credit for U-235	0.0	0.0	-222.8 -222.8	0.0	-222.8
	TOTALS	70.0	333.0	805.8	22.5	1231.4

Table C8.9-5 Annual Cost Summary -- Time Phase E-M

Year	1996	1997	1998	1999	2000	2001	2002	2003	2004	2005	2006	2007	2008	2009	2010	2011	2012	2013	2014	2015	2016	2017
Wet Storage and Transfer																						
RBOF Operations	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	8.2	8.2	8.2	8.2	8.2	8.2	8.2						
L-Basin Operations	17.2	17.2	17.2	17.2	17.2	17.2	17.2	22.2	17.2	12.0	12.0	12.0	12.0	12.0	12.0	12.0						
Subtotal	32.2	32.2	32.2	32.2	32.2	32.2	32.2	37.2	32.2	20.2	20.2	20.2	20.2	20.2	20.2	20.2	0.0	0.0	0.0	0.0	0.0	0.0
Receipt and Treatment Facility																						
Development		4.0	4.0	4.0	8.0	12.0	4.0	4.0														
Implementation					14.3	16.9	65.3	131.4	92.1													
Operations										42.2	42.2	42.2	42.2	42.2	42.2	42.2	20.1	20.1	20.1	20.1	10.1	10.1
Decontam & Decomm																						
Subtotal	0.0	4.0	4.0	4.0	22.3	28.9	69.3	135.4	92.1	42.2	42.2	42.2	42.2	42.2	42.2	42.2	20.1	20.1	20.1	20.1	10.1	10.1
Dry Storage Facility																						
Development																						
Implementation																						
Operations																						
Decontam & Decomm																						
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Glass Log Canisters (incl GWSB Charge)																						
Development																						
Implementation (GWSB Charge)									3.0													
Operations (Canisters)										0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1
Decontam & Decomm																						
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	3.0	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.0	0.0
Adjustments																						
Development																						
Implementation																						
Operations																						
U-235 Credit										-22.3	-22.3	-22.3	-22.3	-22.3	-22.3	-22.3	-11.1	-11.1	-11.1	-11.1	-11.1	-11.1
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	-22.3	-22.3	-22.3	-22.3	-22.3	-22.3	-22.3	-11.1	-11.1	-11.1	-11.1	-11.1	-11.1
Transportation System																						
Development																						
Implementation																						
Operations									2.0	2.0	2.0											
Decontam & Decomm																						
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.0	2.0	2.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Disposal (Repository)																						
Development (Fee)																						
Implementation																				5.0	5.0	5.0
Operations																						
Decontam & Decomm																						
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.0	5.0	5.0	5.0
System Total																						
Development	0.0	4.0	4.0	4.0	8.0	12.0	4.0	4.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.0	5.0	5.0	5.0
Implementation	0.0	0.0	0.0	0.0	14.3	16.9	67.3	133.4	97.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Operations	32.2	32.2	32.2	32.2	32.2	32.2	32.2	37.2	32.2	62.5	62.5	62.5	62.5	62.5	62.5	62.5	20.2	20.2	20.2	20.1	10.1	10.1
Decontam & Decomm	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	-22.3	-22.3	-22.3	-22.3	-22.3	-22.3	-22.3	-11.1	-11.1	-11.1	-11.1	-11.1	-11.1
Subtotal	32.2	36.2	36.2	36.2	54.5	61.1	103.5	174.6	129.3	40.2	40.2	40.2	40.2	40.2	40.2	40.2	9.1	14.1	14.1	14.0	3.9	3.9

Table C8.9-5 Annual Cost Summary -- Time Phase E-M

Year	2018	2019	2020	2021	2022	2023	2024	2025	2026	2027	2028	2029	2030	2031	2032	2033	2034	2035	2036	2037	Total	
Wet Storage and Transfer																						
RBOF Operations																					192.4	
L-Basin Operations																						243.8
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	436.2	
Receipt and Treatment Facility																						
Development																						40.0
Implementation																						320.0
Operations	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1		577.1	
Decontam & Decomm																			11.3	11.3		22.5
Subtotal	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	11.3	11.3		959.7
Dry Storage Facility																						
Development																						0.0
Implementation																						0.0
Operations																						0.0
Decontam & Decomm																						0.0
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Glass Log Canisters (Incl GWSB Charge)																						
Development																						0.0
Implementation (GWSB Charge)																						3.0
Operations (Canisters)																						0.9
Decontam & Decomm																						0.0
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	3.9
Adjustments																						
Development																						0.0
Implementation																						0.0
Operations																						0.0
U-235 Credit																						-222.8
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	-222.8
Transportation System																						
Development																						0.0
Implementation		4.0																				10.0
Operations			0.4	0.4	0.4	0.4	0.4	0.4	0.4	0.4	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1		3.6	
Decontam & Decomm																						0.0
Subtotal	0.0	4.0	0.4	0.4	0.4	0.4	0.4	0.4	0.4	0.4	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.0	0.0	13.6
Disposal (Repository)																						
Development (Fee)	5.0																					30.0
Implementation																						0.0
Operations			1.1	1.1	1.1	1.1	1.1	1.1	1.1	1.1	0.3	0.3	0.3	0.3	0.3	0.3	0.3	0.3	0.3		10.8	
Decontam & Decomm																						0.0
Subtotal	5.0	0.0	1.1	1.1	1.1	1.1	1.1	1.1	1.1	1.1	0.3	0.3	0.3	0.3	0.3	0.3	0.3	0.3	0.3	0.0	0.0	40.8
System Total																						
Development	5.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	70.0
Implementation	0.0	4.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	333.0
Operations	10.1	10.1	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	10.4	10.4	10.4	10.4	10.4	10.4	10.4	10.4	10.4	0.0	0.0	1028.6
Decontam & Decomm	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	11.3	11.3	-200.3
Subtotal	15.1	14.1	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	10.4	10.4	10.4	10.4	10.4	10.4	10.4	10.4	11.3	11.3		1231.4

8.10 Processing

The processing option plans to take maximum opportunity for use of the existing canyons at SRS for processing aluminum-based SNF while costs for these operations can be shared with other programs. As such, the scenario investigated varies from that considered for other technologies so that maximum cost sharing can be achieved. The scenario is evaluated based on a modification of the optimum processing case (Case 2a) detailed in WSRC-RP-95-798, Savannah River Site Evaluation of Spent Nuclear Fuel Options. Operating costs for reprocessing, waste processing and HLW canisters were decreased to reflect the IMNM EIS ROD and the scope of this study (i.e. excluding Mark 16/22s). Processing continues until the end of 2008 when inventories are exhausted. To take advantage of processing, INEL Al-clad fuels are returned to SRS during the period 2000 to 2005. Since INEL deliveries are accelerated, and all FRR shipments are complete in 2008, only a trickle of DRR SNF remains for the period 2009 through 2035. This reduced flow creates the opportunity for selective use of a smaller scale packaging and treatment concept for the remaining period.

The operating concept selected for after 2008 is a small-scale direct disposal (co-disposal) option. A small direct disposal transfer hot cell is integrated into the vault storage facility common to other treatment technologies, as described in section 7.2. The transfer facility is sized for 150 storage tubes, with cost estimates similar to those for the other treatment technologies. Building and equipment costs for the direct transfer hot cell are then added to those costs. The incremental equipment costs for the direct transfer cell are shown in Table 8.10-1.

Table C8.10-1 Additional Equipment for Processing (Direct Disposal) Add-On Option

<u>Equipment</u>	<u>Cost (Thousands of 1996 dollars)</u>
Overhead and Gantry Cranes	1,855
Miscellaneous Equipment	2,747
Allowances	2,770
Total Co-Disposal Transfer Cell	7,372

Additional building space for this transfer module is 1,200 ft² of hot cell space, costed at \$2306 per square foot (1996 dollars), and 400 ft² of makeup and gallery space, costed at \$439 per square foot (1996 dollars), consistent with the transfer facility basis, for an additional building cost of \$2,943,000.

Additional operating and maintenance costs were estimated based on WSRC-TR-95-0180 Life Cycle Cost Estimates for Disposal of Aluminum-Clad HEU SNF (U), scaled to 110% of the melt and dilute option in that study based on higher throughput. The additional annual costs for a 15-shift workweek are shown in Table C8.10-2. Costs for other work patterns were scaled from this value.

Table C8.10-2 Additional Operating and Maintenance Costs for Transfer Module

Item	Cost (Thousands of 1996 dollars)
Labor	1,322
Utilities	36
Materials	5,628
Maintenance Contracts	1,853

Incremental additional cost for decontamination and decommissioning of the transfer module is estimated to be \$4 million.

These costs are added to those of the storage facility sized for the required number of storage locations. Table C8.10-3 presents a summary of the total project costs for the transfer/storage facility with the treatment module included, including design, construction, operation and decommissioning, including appropriate funding overheads. Table C8.10-4 summarizes overall costs for this option, and estimates that approximately 25% of the total new facility cost is associated with storage, and the remaining 75% with transfer activities. Table C8.10-5 shows the distribution of these costs in time

Table C8.10-3 Small Transfer Cell Co-located with Interim Storage Facility

Item	Description	Estimated Cost
DEVELOPMENT COSTS		
0.0	Development	
0.1	Research and Development	5,000
0.2	Process Demonstration	5,000
0.3	Waste Form Qualification	0
0.4	Development Contingency (0% of 0.1, 0.2, and 0.3)	
	SUBTOTAL 0.0 DEVELOPMENT	10,000
IMPLEMENTATION COSTS		
1.0	Construction Costs (Fixed Price Contract)	
1.1	Building Structure Costs	8,206
1.2	Building Equipment Costs	19,227
1.3	Construction Indirects (29% of 1.1 and 1.2)	7,955
	SUBTOTAL Construction 1.0	35,388
2.0	Title I and Title II Design (8% of 1.0)	2,831
3.0	Inspection (3% of 1.0)	1,062
4.0	Project Support (See Note 1) (6% of 1.0)	2,123
5.0	Construction Management (14% of 1.0)	4,954
6.0	SUBTOTAL Engr & Const (1.0, 2.0, 3.0, 4.0, 5.0)	46,358
7.0	Site G & A (16% of 6.0)	7,417
8.0	Total Estimated Cost (TEC) (6.0, 7.0)	53,776
9.0	OPC (Other Project Costs) (See Note 2) (30% of 8.0)	16,133
10.0	Implementation Contingency (0% of 8.0 and 9.0)	
11.0	Total Projected Cost (8.0, 9.0, 10.0) IMPLEMENTATION	69,908
OPERATIONS COSTS		
12.0	Operations and Maintenance Costs (See Note 3)	
12.1	Annual Operation Costs (Labor)	593
12.2	Annual Utility Costs	15
12.3	Annual Material / Consumable Costs	1,881
12.4	Annual Maintenance Costs	1,621
12.5	Interim Storage Costs	Included Above
12.6	Operations Contingency (0% of 12.1 through 12.5)	
	SUBTOTAL 12.0 OPERATIONS (12.1 through 12.6)	4,111
		5 Shifts / Week
		10 Shifts / Week
		15 Shifts / Week
		21 Shifts / Week
		2,246
		61
		7,898
		5,449
		Included Above
		11,325
		15,653
DECONTAMINATION AND DECOMMISSIONING COSTS		
13.0	Decontamination and Decommissioning Costs	
13.1	Facility and Equipment D&D	
13.2	Storage D&D	5942
	SUBTOTAL 13.0 D&D	5942

NOTES:

1. Project Support includes Project Management, Project Control, Estimating and QA
2. Other Project Costs include NEPA process, developing operating procedures, startup and checkout.
3. Annual costs may vary with years depending on operations performed.
4. Percentages shown above are suggested starting point values, and may be adjusted.

Table C8.10-4
Cost Model Processing / Direct Disposal (120 Glass Logs, 300 Co-Disposal Pkgs)

	Cost Items	R&D / Development	Facility & Equipment	Operations	D & D	Total
A	WET STORAGE AND HANDLING - Wet Transfer and Storage	0.0	0.0	434.6 434.6	0.0	434.6
B	TRANSFER AND PACKAGING - New Tran & Stor Facility (Trans and Pkg) - Canisters	10.0 10.0	67.4 52.4 15.0	90.7 83.2 7.5	4.5 4.5	172.6
C	TREATMENT - Canyon Processing Costs - Canyon De-Inventory Costs - DWPF Glass Log Charges	0.0	0.0	643.0 234.4 198.0 210.6	0.0	643.0
D	INTERIM STORAGE - New P&D Facility (Recv, Pkg, and Store) - GWSB Incremental Charge	0.0	20.5 17.5 3.0	27.7 27.7	1.5 1.5	49.7
E	DISPOSAL E1 - Transportation E2 - Repository Fee E3 - Repository Operations	30.0 30.0	10.0 10.0	56.1 11.7 44.4	0.0	96.1
F	CREDITS - Credit for U-235	0.0	0.0	-175.0 -175.0	0.0	-175.0
	TOTALS	40.0	97.9	1077.1	5.9	1221.0

Table C8.10-5 Annual Cost Summary -- Time Phase Repr - DD Co-D

Year	1996	1997	1998	1999	2000	2001	2002	2003	2004	2005	2006	2007	2008	2009	2010	2011	2012	2013	2014	2015	2016	2017
Existing Facilities / Reprocessing																						
RBOF Operations	22.6	15.0	15.0	15.0	18.0	18.0	18.0	18.0	18.0	18.0	15.0	15.0	15.0	15.0	15.0							
L-Basin Operations	11.2	17.2	17.2	17.2	20.2	20.2	20.2	20.2	20.2	20.2												
F & H Canyon Processing	0.0	1.0	12.2	12.2	1.0	3.0	13.0	32.0	32.0	32.0	32.0	32.0	32.0									
Canyon De-inventory Costs																				33.0	33.0	33.0
GWSB Storage Charge									3.0													
Make Glass Logs																						
U-235 Credit					-19.4	-19.4	-19.4	-19.4	-19.4	-19.4	-19.4	-19.4	-19.4									
Subtotal	33.8	33.2	44.4	44.4	19.8	21.8	31.8	50.8	53.8	50.8	27.6	27.6	27.6	15.0	15.0	0.0	0.0	0.0	0.0	33.0	33.0	33.0
Receipt, Treatment & Storage Facility																						
Development						2.5	2.5	2.5	2.5													
Implementation										2.5	18.9	28.4	20.1									
Operations														4.1	4.1	4.1	4.1	4.1	4.1	4.1	4.1	4.1
Decontam & Decomm																						
Subtotal	0.0	0.0	0.0	0.0	0.0	2.5	2.5	2.5	2.5	2.5	18.9	28.4	20.1	4.1	4.1	4.1	4.1	4.1	4.1	4.1	4.1	4.1
Dry Storage Facility																						
Development																						
Implementation																						
Operations																						
Decontam & Decomm																						
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0														
Canisters																						
Development																						
Implementation														5.0	5.0	5.0						
Operations														0.9	0.9	0.9	0.9	0.9	0.9	0.1	0.1	0.1
Decontam & Decomm																						
Subtotal	0.0	5.0	5.0	5.0	0.9	0.9	0.9	0.9	0.9	0.1	0.1	0.1	0.1									
Transportation System																						
Development																						
Implementation														2.0	2.0	2.0						
Operations																						
Decontam & Decomm																						
Subtotal	0.0	2.0	2.0	2.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0									
Disposal (Repository)																						
Development (Fee)																						
Implementation																				5.0	5.0	5.0
Operations																						
Decontam & Decomm																						
Subtotal	0.0	0.0	0.0	0.0	0.0	5.0	5.0	5.0														
System Total																						
Development	0.0	0.0	0.0	0.0	0.0	2.5	2.5	2.5	2.5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.0	5.0	5.0
Implementation	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.5	25.9	35.4	27.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Operations	33.8	33.2	44.4	44.4	19.8	21.8	31.8	50.8	53.8	50.8	27.6	27.6	27.6	20.1	20.1	5.1	5.1	5.1	4.2	37.2	37.2	37.2
Decontam & Decomm	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Subtotal	33.8	33.2	44.4	44.4	19.8	24.3	34.3	53.3	56.3	53.2	53.5	62.9	54.7	20.1	20.1	5.1	5.1	10.1	9.2	42.2	42.2	42.2

Table C8.10-5 Annual Cost Summary -- Time Phase Repr - DD Co-D

Year	2018	2019	2020	2021	2022	2023	2024	2025	2026	2027	2028	2029	2030	2031	2032	2033	2034	2035	2036	2037	Total
Existing Facilities / Reprocessing																					
RBOF Operations																					250.6
L-Basin Operations																					184.0
F & H Canyon Processing																					234.4
Canyon De-inventory Costs	33.0	33.0	33.0																		198.0
GWSB Storage Charge																					3.0
Make Glass Logs						210.6															210.6
U-235 Credit																					-175.0
Subtotal	33.0	33.0	33.0	0.0	0.0	210.6	0.0	905.6													
Receipt, Treatment & Storage Facility																					
Development																					10.0
Implementation																					69.9
Operations	4.1	4.1	4.1	4.1	4.1	4.1	4.1	4.1	4.1	4.1	4.1	4.1	4.1	4.1	4.1	4.1	4.1	4.1			111.0
Decontam & Decomm																			3.0	3.0	5.9
Subtotal	4.1	4.1	4.1	4.1	4.1	4.1	4.1	4.1	4.1	4.1	4.1	4.1	4.1	4.1	4.1	4.1	4.1	4.1	3.0	3.0	196.8
Dry Storage Facility																					
Development																					0.0
Implementation																					0.0
Operations																					0.0
Decontam & Decomm																					0.0
Subtotal	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Canisters																					
Development																					0.0
Implementation																					15.0
Operations	0.1	0.1	0.1	0.1	0.1	1.3	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1				7.5
Decontam & Decomm																					0.0
Subtotal	0.1	0.1	0.1	0.1	0.1	1.3	0.1	0.0	0.0	0.0	22.5										
Transportation System																					
Development																					0.0
Implementation		4.0																			10.0
Operations			1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.2	0.3	0.3	0.3	0.3	0.3	0.3	0.3	0.3			11.7
Decontam & Decomm																					0.0
Subtotal	0.0	4.0	1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.2	0.3	0.0	0.0	21.7							
Disposal (Repository)																					
Development (Fee)	5.0																				30.0
Implementation																					0.0
Operations			4.4	4.4	4.4	4.4	4.4	4.4	4.4	4.4	1.1	1.1	1.1	1.1	1.1	1.1	1.1	1.1			44.4
Decontam & Decomm																					0.0
Subtotal	5.0	0.0	4.4	4.4	4.4	4.4	4.4	4.4	4.4	4.4	1.1	0.0	0.0	74.4							
System Total																					
Development	5.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	40.0
Implementation	0.0	4.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	94.9
Operations	37.2	37.2	42.8	9.8	9.8	221.6	9.8	9.8	9.8	9.8	5.6	5.6	5.6	5.6	5.6	5.6	5.5	5.5	0.0	0.0	1080.1
Decontam & Decomm	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	3.0	3.0	5.9
Subtotal	42.2	41.2	42.8	9.8	9.8	221.6	9.8	9.8	9.8	9.8	5.6	5.6	5.6	5.6	5.6	5.6	5.5	5.5	3.0	3.0	1221.0

9.0 Uncertainty Analyses

Preliminary cost estimates of large, complex, long term projects are inherently uncertain. The projections in this report are no exception. The degree of uncertainty is strongly influenced by the level of maturity of the project in question; some of the technology alternatives evaluated here are very conceptual, and therefore their estimated costs are more uncertain than others. Relative complexity is also a factor; the cost estimates of complex concepts are likely to be more uncertain than those for simpler ones.

In the view of the Team, this relative uncertainty should be taken into account in comparing cost projections for prospective projects of differing maturity and complexity, as in this case. Also, the Team believed it important to present cost estimates in the proper context, making it clear to those who may take action based on recommendations herein that all of the cost figures are uncertain, and some more than others.

To establish a framework for clear and consistent treatment of cost uncertainty, the Team established the following three key cost terms:

Conceptual Estimate

This is the "raw" estimate – the Team's consensus judgment as to the cost of a candidate SNF treatment and disposal option, based on what is currently known about that approach. The conceptual estimate for each option is the compilation of its various cost components, as described earlier in this section. The estimates for each cost component, and therefore the aggregate conceptual estimate, are drawn from experience in comparable applications, from third-party input, from simple estimating calculations, and from the Team members' judgment. Allowances for uncertainties and cost variations are not considered at this stage. These are the costs that were described in detail in Section C8. The conceptual estimates are summarized in Table C9.0-1.

Estimate Range

This is a band of costs, spanning the conceptual estimate and representing the Team's judgement as to the range within which a more complete and better developed cost projection, once available, would likely fall. It reflects the reality that as details develop for any large project, the project's cost estimate inevitable changes, becomes more precise, and usually (but not always) grows. Note that the estimate range, as defined here, is not intended to predict the full range of cost possibilities or to accommodate major project upsets (e.g., licensing delays); rather it is intended to convey the kinds of cost estimate variation which could reasonably be expected as the project proceeds from the conceptual to the preliminary engineering stage.

The first step in the development of the estimated ranges was the review of the conceptual estimate to consider uncertainty. Uncertainty considerations included the source of the cost data and the maturity of the technologies. The conceptual estimate for each technology was assessed to assign a negative (-%) and a positive (+%) variance for each of the cost components. For example, the variance for the wet storage and handling costs was determined to be -10% to +30%

for all technologies, as the costs are based upon historical data and the type of activities performed are generally the same for all the technologies. On the other hand, the variance assigned to the treatment costs varied more widely (-40% to +100%) among the technologies due to a significant difference in maturity. The cost variance percentages by technology and cost component are presented in Table C9.0-2.

The second and final step was to compute the total composite estimate range for each option. This was done by applying the negative variances to the conceptual estimate to calculate the low range cost and the positive variances to calculate the high range cost for each option. The cost ranges are shown in Figure C5.0-1 and Table C9.0-3.

Comparative Cost

Comparative costs were developed to reflect the relative differences in uncertainty among the options. Based upon the cost variances for each technology and cost component, discussed above, relative uncertainty factors (RUF) were derived by assessing the difference in positive variance. For example, the RUF for the wet storage and handling costs, was 0% for all technologies, as the positive variance (+30%) is the same for all technologies. On the other hand, the RUF for treatment costs ranges from +0% for Processing, where uncertainties are small, to +90% for the advanced technologies, where uncertainties are significantly greater. The RUF by technology and cost component are presented in Table C9.0-2. Once the RUFs were defined, the comparative costs were calculated by applying these factors to the appropriate component of the conceptual estimate (Table C9.0-1), then summing to arrive at the total comparative cost for each technology. The product of the RUF and the conceptual cost component, which is the uncertainty component of the comparative cost, is shown in Table C9.0-4. The comparative costs are presented in Table C5.0-2.

In order to include the uncertainty component of the comparative cost in the cash flow estimates described in Section C8, the uncertainty component was distributed over time. The uncertainty component of Transfer and Packaging for each technology was assumed to be spread evenly over the three years prior to startup of the treatment phase. The uncertainty component of treatment was assumed to be spread evenly over the six years beginning with the startup of the treatment module. Finally the uncertainty component of interim storage was assumed to be spread evenly over the years 2020 to 2027. This is because these costs are related to the need to vent the containers prior to shipping to the repository. The conceptual cost cash flow by year, the uncertainty cash flow by year and the total comparative cost cash flow by year are shown for each technology in Tables C9.0-5,1-3.

Summary

The objective of the cost evaluation was to consistently compare costs among the technologies using available pre-conceptual design information. The approach used to develop the estimate and apply uncertainties reflects the application of considerable experience and judgement by the Team.

TABLE C9.0--1 SUMMARY OF CONCEPTUAL COST INFORMATION (Millions of 1996 Dollars¹)

Option	Wet Storage and Handling	Transfer and Packaging	Treatment	Interim Storage	Disposal			Adjustments ²	Total
					Transportation	Repository Fee	Repository Operations		
Direct Disposal (1100 Small Packages)	282	440	0	120	37	30	368	61	1338
Direct Disposal - CoDisposal (1400 Co-Disposal Pkgs)	282	427	0	126	42	30	140	61	1108
Press and Dilute (400 Co-Disposal Pkgs - 20%)	347	417	231	97	19	30	40	61	1242
Press and Dilute (1300 Co-Disposal Pkgs - 2%)	347	436	231	122	40	30	130	61	1396
Melt and Dilute (400 Co-Disposal Pkgs)	347	392	274	97	19	30	40	0	1199
Plasma Arc (400 Co-Disposal Pkgs)	456	380	453	93	19	30	40	0	1472
GMODS (800 Co-Disposal Pkgs)	456	388	407	106	28	30	80	0	1496
Dissolve and Vitrify (800 Co-Disposal Pkgs)	456	388	718	106	28	30	80	0	1806
Electrometallurgical (90 Glass Logs)	436	364	596	3	14	30	11	-223	1231
Processing / Direct Disposal (120 Glass Logs, 300 Co-Disposal Pkgs)	435	173	643	50	22	30	44	-175	1221

Note 1: The estimates presented above represent a comparative best estimate for the values shown, without contingencies.

Note 2: Adjustments include charges for Hot Vacuum Drying Facility and credits for sale of U-235 when appropriate.

Table C9.0-2 Cost Estimate Range Assumptions

Option	Wet Storage and Handling			Transfer and Packaging			Treatment			Interim Storage			Disposal									Adjustments ²		
	-	RUF	+	-	RUF	+	-	RUF	+	-	RUF	+	Transportation			Repository Fee			Repository Operations			-	RUF	+
Direct Disposal (1100 Small Packages)	10%	0%	30%	10%	0%	30%	10%	10%	20%	10%	20%	40%	10%	0%	20%	0%	0%	0%	10%	0%	30%	10%	0%	30%
Direct Disposal - CoDisposal (1400 Co-Disposal Pkgs)	10%	0%	30%	10%	10%	40%	10%	10%	20%	10%	20%	40%	10%	0%	20%	0%	0%	0%	10%	0%	30%	10%	0%	30%
Press and Dilute (400 Co-Disposal Pkgs - 20%)	10%	0%	30%	10%	10%	40%	10%	40%	50%	10%	20%	40%	10%	0%	20%	0%	0%	0%	10%	0%	30%	10%	0%	30%
Press and Dilute (1300 Co-Disposal Pkgs 2%)	10%	0%	30%	10%	10%	40%	10%	40%	50%	10%	20%	40%	10%	0%	20%	0%	0%	0%	10%	0%	30%	10%	0%	30%
Melt and Dilute (400 Co-Disposal Pkgs)	10%	0%	30%	10%	10%	40%	10%	40%	50%	10%	0%	20%	10%	0%	20%	0%	0%	0%	10%	0%	30%	0%	0%	0%
Plasma Arc (400 Co-Disposal Pkgs)	10%	0%	30%	10%	10%	40%	0%	90%	100%	10%	0%	20%	10%	0%	20%	0%	0%	0%	10%	0%	30%	0%	0%	0%
GMODS (800 Co-Disposal Pkgs)	10%	0%	30%	10%	10%	40%	0%	90%	100%	10%	0%	20%	10%	0%	20%	0%	0%	0%	10%	0%	30%	0%	0%	0%
Dissolve and Vitrify (800 Co-Disposal Pkgs)	10%	0%	30%	10%	10%	40%	0%	20%	30%	10%	0%	20%	10%	0%	20%	0%	0%	0%	10%	0%	30%	0%	0%	0%
Electrometallurgical (90 Glass Logs)	10%	0%	30%	10%	0%	30%	0%	60%	70%	10%	0%	20%	10%	0%	20%	0%	0%	0%	10%	0%	30%	10%	0%	30%
Processing / Direct Disposal (120 Glass Logs, 300 Co-Disposal Pkgs)	10%	0%	30%	10%	0%	30%	43%	0%	10%	10%	20%	40%	10%	0%	20%	0%	0%	0%	10%	0%	30%	10%	0%	30%

RUF = Relative Uncertainty Factor

Table C9.0-3 Cost Estimate Ranges

Option	Minimum	Conceptual	Comparison	Maximum
0	1207	990	1014	1739
Total System Cost Direct Disposal - CoDisposal (1400 Co-Disposal Pkgs)	1000	1108	1176	1482
Task Team Cost Model (\$Millions) Press and Dilute (400 Co-Disposal Pkgs - 20%)	1120	1242	1395	1701
Task Team Cost Model (\$million) Press and Dilute (1300 Co-Disposal Pkgs - 2%)	1260	1396	1556	1904
Task Team Cost Model (\$million) Melt and Dilute (400 Co-Disposal Pkgs)	1082	1199	1348	1632
Task Team Cost Model (\$million) Plasma Arc (400 Co-Disposal Pkgs)	1373	1472	1918	2249
Task Team Cost Model (\$million) GMODS (800 Co-Disposal Pkgs)	1390	1496	1902	2247
Task Team Cost Model (\$million) Dissolve and Vitrify (800 Co-Disposal)	1700	1806	1989	2365
Task Team Cost Model (\$million) Electrometallurgical (90 Glass Logs)	1171	1231	1625	1865
Task Team Cost Model Processing / Direct Disposal (120 Glass Logs, 300 Co-Disposal Pkgs)	888	1221	1231	1452

Table C9.0-4 Uncertainty Component of Comparative Cost

Option	Wet Storage and Handling	Transfer and Packaging	Treatment	Interim Storage	Disposal			Adjustments ²	Total
					Transportation	Repository Fee	Repository Operations		
Direct Disposal (1100 Small Packages)	0	0	0	24	0	0	0	0	24
Direct Disposal - CoDisposal (1400 Co-Disposal Pkgs)	0	43	0	25	0	0	0	0	68
Press and Dilute (400 Co-Disposal Pkgs - 20%)	0	42	92	19	0	0	0	0	153
Press and Dilute (1300 Co-Disposal Pkgs - 2%)	0	44	92	24	0	0	0	0	160
Melt and Dilute (400 Co-Disposal Pkgs)	0	39	110	0	0	0	0	0	149
Plasma Arc (400 Co-Disposal Pkgs)	0	38	408	0	0	0	0	0	446
GMODS (800 Co-Disposal Pkgs)	0	39	367	0	0	0	0	0	406
Dissolve and Vitrify (800 Co-Disposal Pkgs)	0	39	144	0	0	0	0	0	182
Electrometallurgical (90 Glass Logs)	0	0	358	0	0	0	0	0	358
Processing / Direct Disposal (120 Glass Logs, 300 Co-Disposal Pkgs)	0	0	0	10	0	0	0	0	10

**Table C9.0-5 Cash Flows
Sheet 1**

Conceptual Costs

Year	1996	1997	1998	1999	2000	2001	2002	2003	2004	2005	2006	2007	2008	2009	2010	2011	2012	2013	2014	2015	2016	
Direct Disposal	32.2	32.2	32.2	32.2	32.2	40.5	45.5	50.5	50.5	40.5	40.5	10.0	10.0	10.0	10.0	10.0	10.0	10.0	4.4	4.4	4.4	4.4
Co-Disposal	32.2	34.2	32.4	141.4	126.9	53.9	55.4	49.3	49.3	39.3	39.3	9.4	9.4	9.4	9.4	9.4	9.4	9.4	9.4	9.4	9.4	9.4
Press and Dilute (20%)	32.2	34.5	34.5	45.2	109.3	140.5	118.9	60.7	62.2	58.1	58.1	48.1	48.1	13.4	13.4	13.4	13.4	13.4	18.4	18.4	18.4	11.8
Press and Dilute (2%)	32.2	34.5	34.5	45.8	114.0	147.5	121.9	62.9	64.4	60.3	60.3	50.3	50.3	14.4	14.4	14.4	14.4	14.4	19.4	19.4	19.4	12.1
Melt and Dilute	32.2	35.2	35.2	47.5	118.6	148.2	115.1	50.8	50.8	50.8	50.8	50.8	50.8	14.7	14.7	14.7	14.7	14.7	19.7	19.7	19.7	12.5
Plasma Arc	32.2	37.2	37.2	37.2	42.2	66.2	59.7	136.0	213.8	149.6	50.8	50.8	50.8	50.8	50.8	50.8	50.8	50.8	19.8	19.8	19.8	12.5
GMODS	32.2	37.2	37.2	37.2	42.2	64.6	57.7	128.5	199.7	139.0	51.8	51.8	51.8	51.8	51.8	51.8	51.8	51.8	20.3	20.3	20.3	12.6
Dissolve and Vitrify	32.2	34.2	34.2	34.2	36.2	61.7	62.0	153.4	254.8	178.4	67.2	67.2	67.2	67.2	67.2	67.2	67.2	67.2	27.7	27.7	27.7	16.2
Electrometallurgical	32.2	36.2	36.2	36.2	54.5	61.1	103.5	174.6	129.3	40.2	40.2	40.2	40.2	40.2	40.2	40.2	40.2	9.1	14.1	14.1	14.0	3.9
Processing (Baseline)	33.8	33.2	44.4	44.4	19.8	24.3	34.3	53.3	56.3	53.2	53.5	62.9	54.7	20.1	20.1	5.1	5.1	10.1	9.2	42.2	42.2	42.2

Relative Uncertainty

Year	1996	1997	1998	1999	2000	2001	2002	2003	2004	2005	2006	2007	2008	2009	2010	2011	2012	2013	2014	2015	2016	
Direct Disposal			0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0											
Co-Disposal			14.2	14.2	14.2	0.0	0.0	0.0	0.0	0.0	0.0											
Press and Dilute (20%)					13.9	13.9	13.9	15.4	15.4	15.4	15.4	15.4	15.4									
Press and Dilute (2%)					14.5	14.5	14.5	15.4	15.4	15.4	15.4	15.4	15.4									
Melt and Dilute					13.1	13.1	13.1	18.3	18.3	18.3	18.3	18.3	18.3									
Plasma Arc								12.7	12.7	12.7	67.9	67.9	67.9	67.9	67.9	67.9						
GMODS								12.9	12.9	12.9	61.1	61.1	61.1	61.1	61.1	61.1						
Dissolve and Vitrify								12.9	12.9	12.9	23.9	23.9	23.9	23.9	23.9	23.9						
Electrometallurgical							12.1	12.1	12.1	59.6	59.6	59.6	59.6	59.6	59.6							
Processing (Baseline)										0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

Total System Costs

Year	1996	1997	1998	1999	2000	2001	2002	2003	2004	2005	2006	2007	2008	2009	2010	2011	2012	2013	2014	2015	2016	
Direct Disposal	32.2	32.2	32.2	32.2	32.2	40.5	45.5	50.5	50.5	40.5	40.5	10.0	10.0	10.0	10.0	10.0	10.0	10.0	4.4	4.4	4.4	4.4
Co-Disposal	32.2	34.2	106.6	155.6	141.1	53.9	55.4	49.3	49.3	39.3	39.3	9.4	9.4	9.4	9.4	9.4	9.4	9.4	9.4	9.4	9.4	9.4
Press and Dilute (20%)	32.2	34.5	34.5	45.2	123.2	154.4	130.8	76.0	77.5	73.4	73.4	63.4	63.4	13.4	13.4	13.4	13.4	13.4	18.4	18.4	18.4	11.8
Press and Dilute (2%)	32.2	34.5	34.5	45.8	128.5	162.1	136.4	78.3	79.8	75.7	75.7	65.7	65.7	14.4	14.4	14.4	14.4	14.4	19.4	19.4	19.4	12.1
Melt and Dilute	32.2	35.2	35.2	47.5	129.7	161.3	128.1	69.0	69.0	69.0	69.0	69.0	69.0	14.7	14.7	14.7	14.7	14.7	19.7	19.7	19.7	12.5
Plasma Arc	32.2	37.2	37.2	37.2	42.2	66.2	59.7	148.7	226.5	162.3	118.7	118.7	118.7	118.7	118.7	118.7	50.8	50.8	19.8	19.8	19.8	12.5
GMODS	32.2	37.2	37.2	37.2	42.2	64.6	57.7	141.4	211.6	152.0	112.9	112.9	112.9	112.9	112.9	112.9	51.8	51.8	20.3	20.3	20.3	12.6
Dissolve and Vitrify	32.2	34.2	34.2	34.2	36.2	61.7	62.0	166.3	267.8	191.4	91.1	91.1	91.1	91.1	91.1	91.1	67.2	67.2	27.7	27.7	27.7	16.2
Electrometallurgical	32.2	36.2	36.2	36.2	54.5	61.1	115.6	188.7	141.5	99.8	99.8	99.8	99.8	99.8	99.8	99.8	40.2	40.2	14.1	14.1	14.0	3.9
Processing (Baseline)	33.8	33.2	44.4	44.4	19.8	24.3	34.3	53.3	56.3	53.2	53.5	62.9	54.7	20.1	20.1	5.1	5.1	10.1	9.2	42.2	42.2	42.2

**Table C9.0-5 Cash Flows
Sheet 2**

Conceptual Costs

Year	2017	2018	2019	2020	2021	2022	2023	2024	2025	2026	2027	2028	2029	2030	2031	2032	2033	2034	2035	2036	2037	Total		
Direct Disposal	4.4	4.4	4.4	44.5	44.5	44.5	44.5	44.5	44.5	44.5	44.5	14.3	14.3	14.3	14.3	14.3	14.3	14.3	14.3	14.3	0.0	0.0	990.4	
Co-Disposal	9.4	9.4	8.4	22.2	22.2	22.2	22.2	22.2	22.2	22.2	22.2	8.7	8.7	8.7	8.7	8.7	8.7	8.7	8.7	8.7	13.1	13.1	1107.9	
Press and Dilute (20%)	11.8	11.8	10.8	18.1	18.1	18.1	18.1	18.1	18.1	18.1	18.1	8.1	8.1	8.1	8.1	8.1	8.1	8.1	8.1	8.1	14.5	14.5	1241.5	
Press and Dilute (2%)	12.1	12.1	11.1	29.4	29.4	29.4	29.4	29.4	29.4	29.4	29.4	11.1	11.1	11.1	11.1	11.1	11.1	11.1	11.1	11.1	15.0	15.0	1396.1	
Melt and Dilute	12.5	12.5	11.5	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	8.7	8.7	8.7	8.7	8.7	8.7	8.7	8.7	8.7	12.0	12.0	1199.1	
Plasma Arc	12.5	12.5	11.5	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	8.7	8.7	8.7	8.7	8.7	8.7	8.7	8.7	8.7	12.0	12.0	1472.3	
GMODS	12.6	12.6	11.6	18.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	12.2	12.2	1496.2
Dissolve and Vitrify	16.2	16.2	15.2	21.7	21.7	21.7	21.7	21.7	21.7	21.7	21.7	13.7	13.7	13.7	13.7	13.7	13.7	13.7	13.7	13.7	13.7	12.2	12.2	1806.4
Electrometallurgical	3.9	15.1	14.1	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	10.4	10.4	10.4	10.4	10.4	10.4	10.4	10.4	10.4	11.3	11.3	1231.4	
Processing (Baseline)	42.2	42.2	41.2	42.8	9.8	9.8	221.6	9.8	9.8	9.8	9.8	5.6	5.6	5.6	5.6	5.6	5.6	5.6	5.6	5.5	3.0	3.0	1221.0	

Relative Uncertainty

Year	2017	2018	2019	2020	2021	2022	2023	2024	2025	2026	2027	2028	2029	2030	2031	2032	2033	2034	2035	2036	2037	Total		
Direct Disposal				3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0												24.0	
Co-Disposal				3.2	3.2	3.2	3.2	3.2	3.2	3.2	3.2													67.9
Press and Dilute (20%)				2.4	2.4	2.4	2.4	2.4	2.4	2.4	2.4													153.4
Press and Dilute (2%)				3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1													160.3
Melt and Dilute				0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0													148.8
Plasma Arc				0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0													445.7
GMODS				0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0													405.5
Dissolve and Vitrify				0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0													182.3
Electrometallurgical				0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0													394.1
Processing (Baseline)				1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.2													9.8

Total System Costs

Year	2017	2018	2019	2020	2021	2022	2023	2024	2025	2026	2027	2028	2029	2030	2031	2032	2033	2034	2035	2036	2037	Total		
Direct Disposal	4.4	4.4	4.4	47.5	47.5	47.5	47.5	47.5	47.5	47.5	47.5	14.3	14.3	14.3	14.3	14.3	14.3	14.3	14.3	14.3	0.0	0.0	1014.5	
Co-Disposal	9.4	9.4	8.4	25.4	25.4	25.4	25.4	25.4	25.4	25.4	25.4	8.7	8.7	8.7	8.7	8.7	8.7	8.7	8.7	8.7	13.1	13.1	1175.8	
Press and Dilute (20%)	11.8	11.8	10.8	20.5	20.5	20.5	20.5	20.5	20.5	20.5	20.5	8.1	8.1	8.1	8.1	8.1	8.1	8.1	8.1	8.1	14.5	14.5	1394.9	
Press and Dilute (2%)	12.1	12.1	11.1	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	11.1	11.1	11.1	11.1	11.1	11.1	11.1	11.1	11.1	15.0	15.0	1556.4	
Melt and Dilute	12.5	12.5	11.5	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	8.7	8.7	8.7	8.7	8.7	8.7	8.7	8.7	8.7	12.0	12.0	1347.9	
Plasma Arc	12.5	12.5	11.5	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	8.7	8.7	8.7	8.7	8.7	8.7	8.7	8.7	8.7	12.0	12.0	1918.0	
GMODS	12.6	12.6	11.6	18.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	12.2	12.2	1901.7
Dissolve and Vitrify	16.2	16.2	15.2	21.7	21.7	21.7	21.7	21.7	21.7	21.7	21.7	13.7	13.7	13.7	13.7	13.7	13.7	13.7	13.7	13.7	13.7	12.2	12.2	1988.7
Electrometallurgical	3.9	15.1	14.1	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	10.4	10.4	10.4	10.4	10.4	10.4	10.4	10.4	10.4	11.3	11.3	1625.9	
Processing (Baseline)	42.2	42.2	41.2	44.0	11.0	11.0	222.8	11.0	11.0	11.0	11.0	5.6	5.6	5.6	5.6	5.6	5.6	5.6	5.5	5.5	3.0	3.0	1230.9	

10.0 Variations Considered

The Cost Team evaluated several variations on the operating concept process flow for each technology. These variations are still quite conceptual in nature and have been reviewed to a varying extent, depending on the time and resources available to the team. Cost evaluations were performed to the degree practical to identify operating concepts which merit further study in the near future.

10.1 Point of Origin Packaging - SNF loaded into interim storage or disposal packages at the reactor sites

An operating concept with considerable potential for overall system economy is the loading of direct disposal packages at the reactor sites. SNF would be loaded directly into the small-diameter co-disposal packages at the domestic or foreign research reactors. This would significantly reduce work required at SRS, since loaded packages could be received and transferred directly to on-site interim dry storage. If it is cost effective to also use the transportation cask for interim dry storage, only a concrete pad and off-loading equipment would be required at SRS. The operation of such systems would be very similar to commercial canister-based systems used by several vendors for storage and transportation. Loading, welding, and drying operations would be performed at the sending site by SRS personnel or by crews trained by SRS personnel.

A similar concept would be the use of dual-purpose canisters or casks, also loaded directly at the sending reactor site. In this case, the system would only be designed for interim storage and transportation and would not be designed to interface with the repository overpack. Co-disposal packages would not be used, and the fuel assemblies would be loaded directly into the cask. Although the repository limits on the quantity of fissile material in each package would not be applicable, small cask systems would likely be used to match handling limitations at the reactors. This concept would also minimize the operations required at SRS, since the casks would be stored on a concrete pad until they are transported to the repository. At the repository, the SNF would be repackaged into a direct disposal package, since the storage and transportation cask was not designed for the disposal function. This concept would also postpone the decision to determine what eventually will, or will not, be determined acceptable for direct inclusion in the repository.

Both these concepts are related to direct disposal methods. Cost impacts for these alternatives have not been explicitly evaluated. If adopted, no construction or operation of new facilities would be required, and therefore savings could be significant. There will likely be significant difficulties in implementing these options, since they do place many requirements on the sending reactor sites. These requirements may not be enforceable for all sites, and unless there is almost universal implementation of this option, there may not be enough impact to change overall system construction and operation requirements. In addition, a system which requires repackaging at the repository does not satisfy the overall working guidelines established for this task force. This option should be explored to determine if it represents an alternative which could reasonably be implemented.

10.2 Use existing basins in lieu of building a new dry transfer facility

Another alternative operating concept would involve using RBOF and L-Basin to transfer SNF into the direct disposal packages. This would avoid construction costs for a new facility, at the expense of continuing the operating costs for these wet basins. This concept would require construction of a Cold Vacuum Drying Facility (CVD) to dry and seal the packages. This CVD Facility would be similar to the one to be constructed at Hanford for the N Reactor SNF dry storage project. In addition, if uranium metal fuel is also to be accommodated, the CVD Facility would require a major addition designed for hot vacuum drying, similar to the Hanford HVD Facility for N Reactor SNF.

The operating flow for this concept is shown in Figure C10.2-1. Package loading operations would be performed in the cask pit area of the pool or in adjacent non-critical path pool areas to enhance throughput. Once the loaded package is placed in the cask, the package shield plug is set in place, and the cask lid installed and bolted. The unsealed package is then moved in a vertical position in the sealed cask to the CVD Facility. At the CVD Facility, draining, welding and drying operations would be performed off critical path, then the package transferred to the interim storage facility to await shipment to the repository.

An analysis was made of this option, comparing it to the direct disposal method since direct disposal package dimensions are best suited to loading in the basin transfer areas. The team used the assembly and cask throughput assumptions of Table C3.0-1 and Table C3.0-2, and the material handling assumptions shown in Table C3.0-3. The material handling assumptions are based on historical durations, and assumptions about system improvements which can reasonably be expected, such as increasing the payload of the on-site CD casks.

The Cost Team calculated the time required to receive off-site shipments for each year, and to trans-ship between RBOF and L-basin for those casks which do not fit in the L-basin receiving area. Any remaining time during the year was allocated to wet loading of direct disposal packages. It was assumed that RBOF loads direct disposal packages holding 36 MTRE assemblies, and L-basin loads direct disposal packages containing 24 assemblies. It was further assumed, based on SRS experience, that 16 work shifts are required to deliver, load remove each direct disposal package from the basin transfer area. In this manner, the backlog of assemblies stored in the basins is worked off as available time permits. This analysis showed that at full utilization, L-basin would work 21 shifts per week until it is de-inventoried in 2021. RBOF operations could be scaled back to day-shift-only in 2017, but that facility would have to remain in operation until 2035. Based on estimates for Hanford SNF, the CVD facility is estimated at \$25 million capital cost, \$4 million per year operating costs, and \$3 million D&D costs. Table C10.2-1 summarizes the costs associated with this option. The total system cost is \$1740 million compared to \$1,362 million for the direct disposal option. On this basis, this alternative does not appear cost effective due to increased costs resulting from additional operational time for the existing basins.

Potential enhancements to improve through-put in the existing basins, which deserve further consideration, include productivity improvements and use of additional cask pits. The current through-put estimate is based upon historical handling durations for cask receipts in a

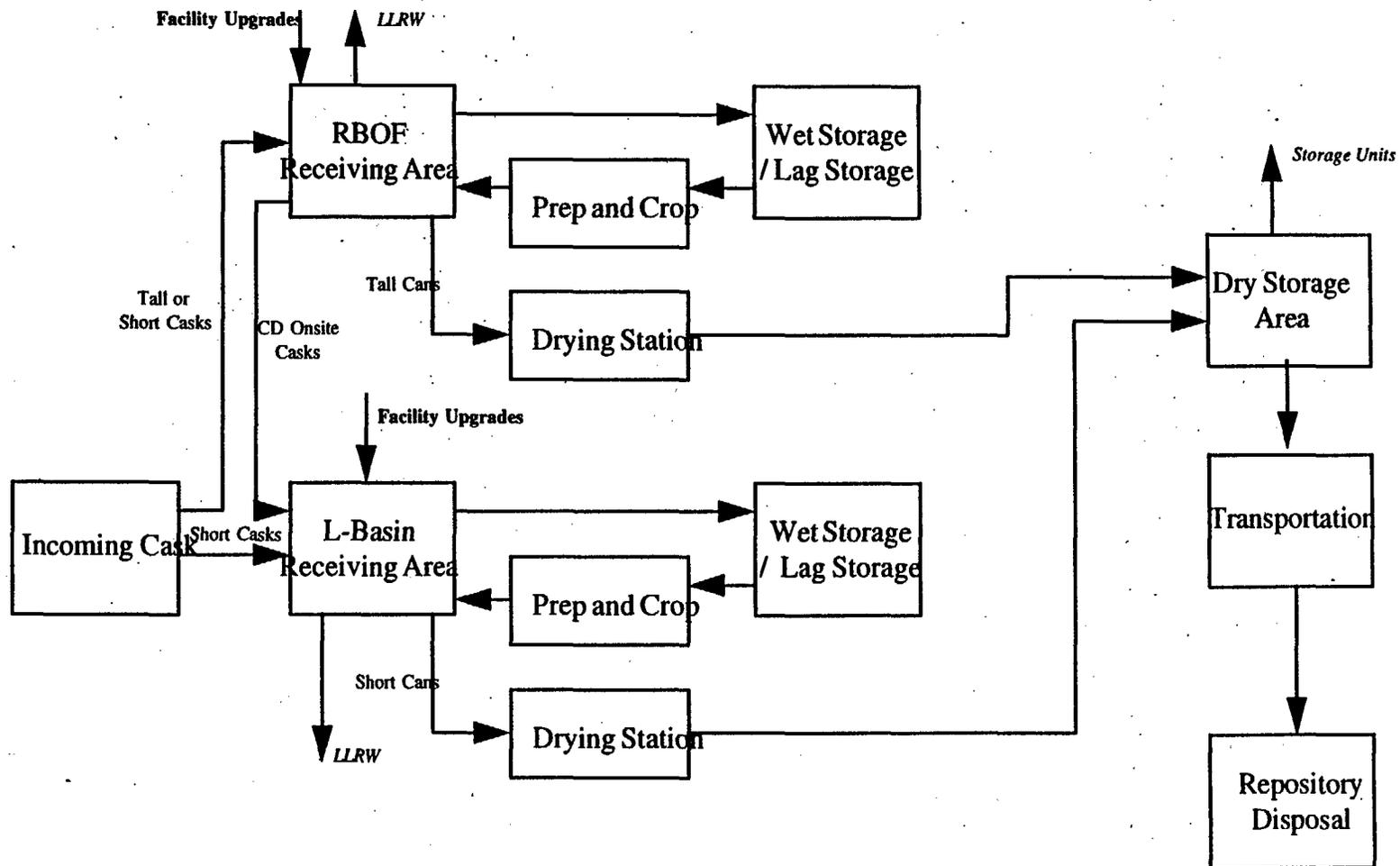


Figure C10.2-1 Operating Flow for "No New Facility" Concept

**Table C10.2-1
Direct Disposal (1100 Small Packages) NO NEW FACILITIES**

	Cost Items	Front End (R&D/Dev)	Facility & Equip	Operations (Unit Opns)	Back End (D&D)	Total
A	Front End Storage Cost	0.0	15.0	960.7	0.0	975.7
	- RBOF			471.0		
	- L-Basin			447.0		
	- Canisters		15.0	42.7		
B	Treatment	0.0	56.2	145.0	8.0	209.2
	- HVD Facility		31.2	25.0	5.0	
	- CVD Facility		25.0	120.0	3.0	
C	Packaging (with drying) and Onsite Handling	0.0	0.0	0.0	0.0	0.0
	- Transfer Facility					
	- Canisters					
D	Interim Storage	0.0	56.0	61.8	2.3	120.1
	- Vault Storage		56.0	61.8	2.3	
E	Repository	30.0	0.0	0.0	0.0	30.0
	- Fee	30.0				
F	Repository	0.0	0.0	368.1	0.0	368.1
	- Operations			368.1		
G	Transport to Repository	0.0	10.0	26.7	0.0	36.7
	- Cask Development & Purchase		10.0			
	- Transportation Operations			26.7		
	TOTALS	30.0	137.2	1562.2	10.3	1739.8

Notes: Includes \$61 million for U-metal Fuel HVD Facility.

non-production-infrequent receipt mode. The cask handling duration could possibly be improved considering the production mode rate required to handle the FRR SNF. In addition, moving the package sealing operations to the CVD facility could free up some critical path time in the basins.

A significant factor to be considered in a decision to use the existing basins is their condition and age. While they are currently in good condition, the 2021 and 2035 cease operation dates, respectively, for L-Basin and RBOF is a significant demand on these aging structures.

10.3 Expedite startup of the transfer facility

This alternative would expedite closure of the existing basins by proceeding with development and construction of the new dry transfer and interim storage facilities as soon as possible, with provisions to add on the treatment module at a later date. Initially, the SNF would be transferred from the existing basins or received from shipping reactors, placed in a direct disposal package, then temporarily placed in the storage facility. Once the treatment module started operation, the temporarily stored SNF would be removed from the storage facility, treated, repackaged (if needed), and placed in interim storage awaiting transport to the repository.

An analysis was made of the flow of SNF assemblies from the basins to the transfer facility, to the interim storage facility, back to the transfer facility, through treatment, and back to interim storage. The co-disposal option typically requires more packages than most treatment options. Because of this, the analysis had the objective of ensuring that the transfer facility was fully utilized, and that SNF was removed from temporary packaging as soon as possible. Three cases were analyzed, one with a relatively late final treatment implementation (GMODS in 2006), one with a relatively early treatment implementation (Melt and Dilute in 2003) and EM (start in 2005). EM was considered because no new storage facility would be required and therefore EM is the technology that should reflect a worst case scenario for early startup of the transfer and storage facility. Comparable differences would be expected for treatment technologies with similar implementation dates to the first two cases. The differences in operation between the base case for the three options, and this revised scenario were evaluated. Figure C10.3-1, C10.3-2 and C10.2-3 show the comparison of the operations of the alternate concepts.

For the cost comparison, it was assumed that co-disposal canisters used initially for direct disposal packaging could be re-used for the final treatment technology, and that no additional costs for canisters were required. Additionally, it was assumed that there would be no significant change in the number of interim storage locations required, so the facility costs would be constant for the transfer and storage facilities. It was assumed that the cost of building the treatment module would be increased by 100% of the building costs, and 25% of the equipment costs because of the difficulty involved in constructing nuclear facilities in phases. Early implementation of the transfer and storage facilities would permit decommissioning of RBOF and L-basin two years early for the Melt and Dilute option, five years early for EM and six years early for the GMODS option. The remaining differences in cost are a result of the different years of operations of the transfer and storage facility. Tables C10.3-1 and C10.3-2 summarize the cost savings expected for the late (GMODS) technology and the early (Melt and Dilute) technology, respectively.

Figure C10.3-1 Comparison of Shift Requirements - GMODS

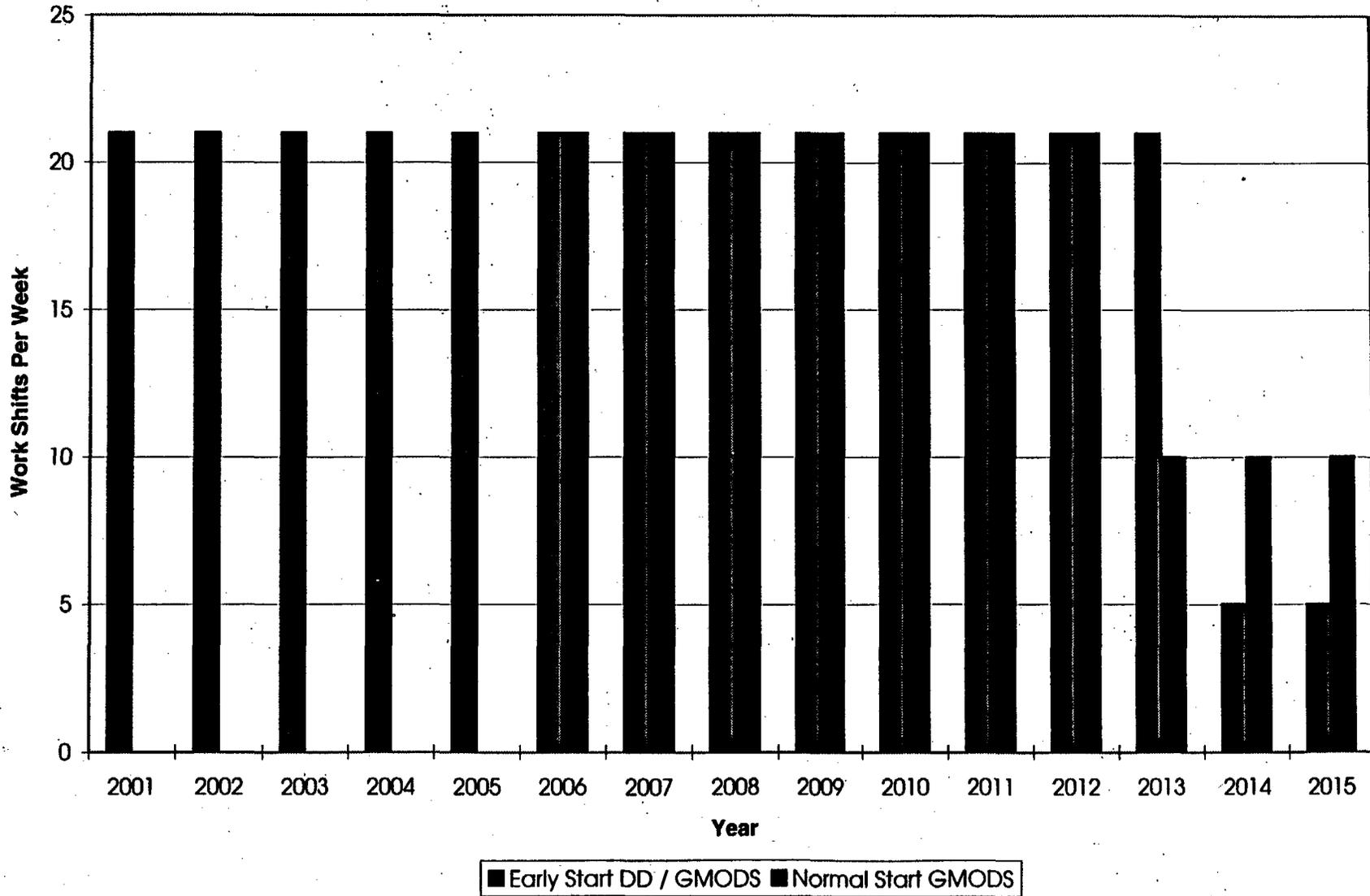


Figure C10.3-2 Comparison of Shift Requirements- Melt and Dilute

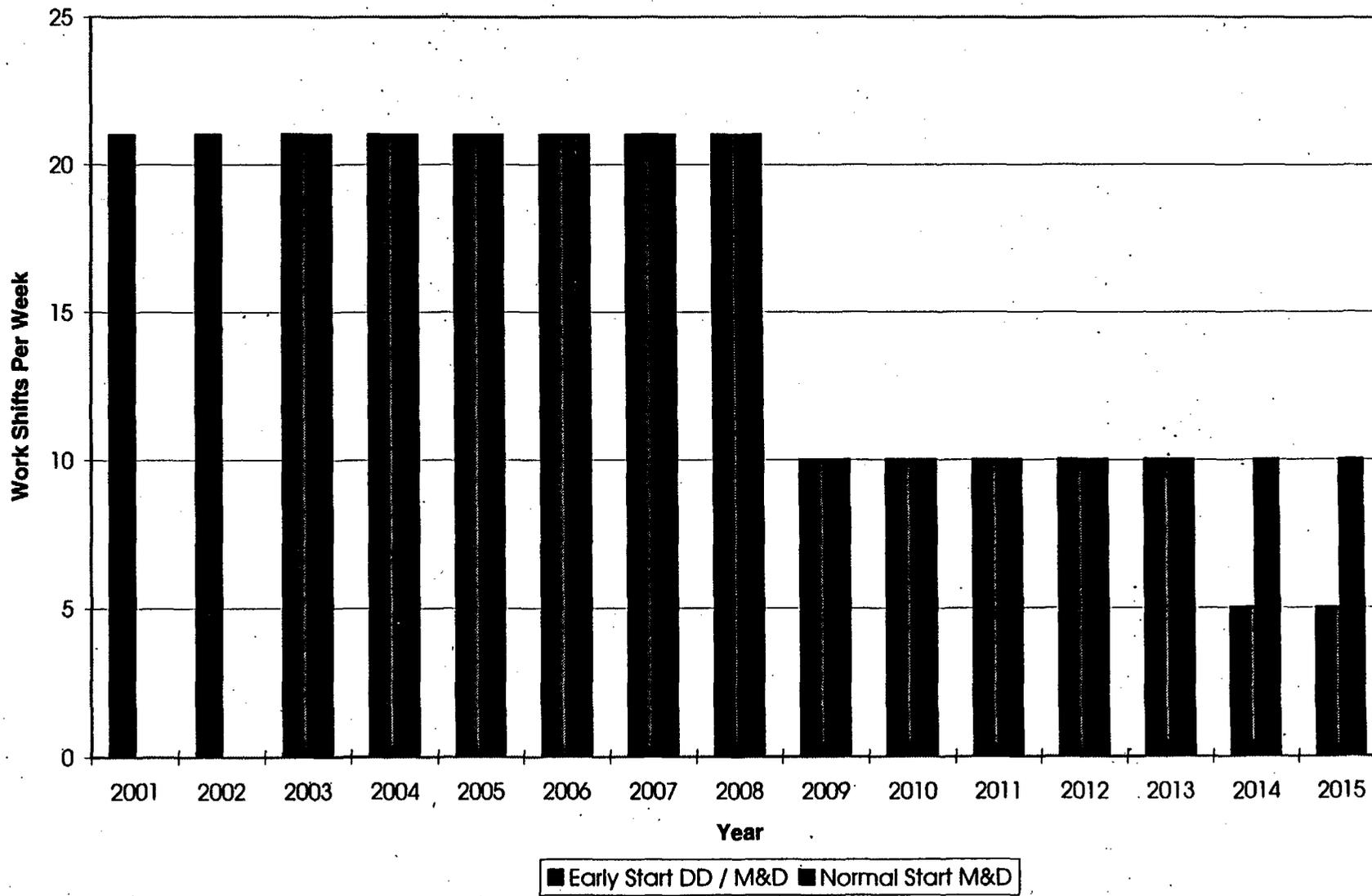


Figure C10.3-3 Comparison of Shift Requirements - Electrometallurgical

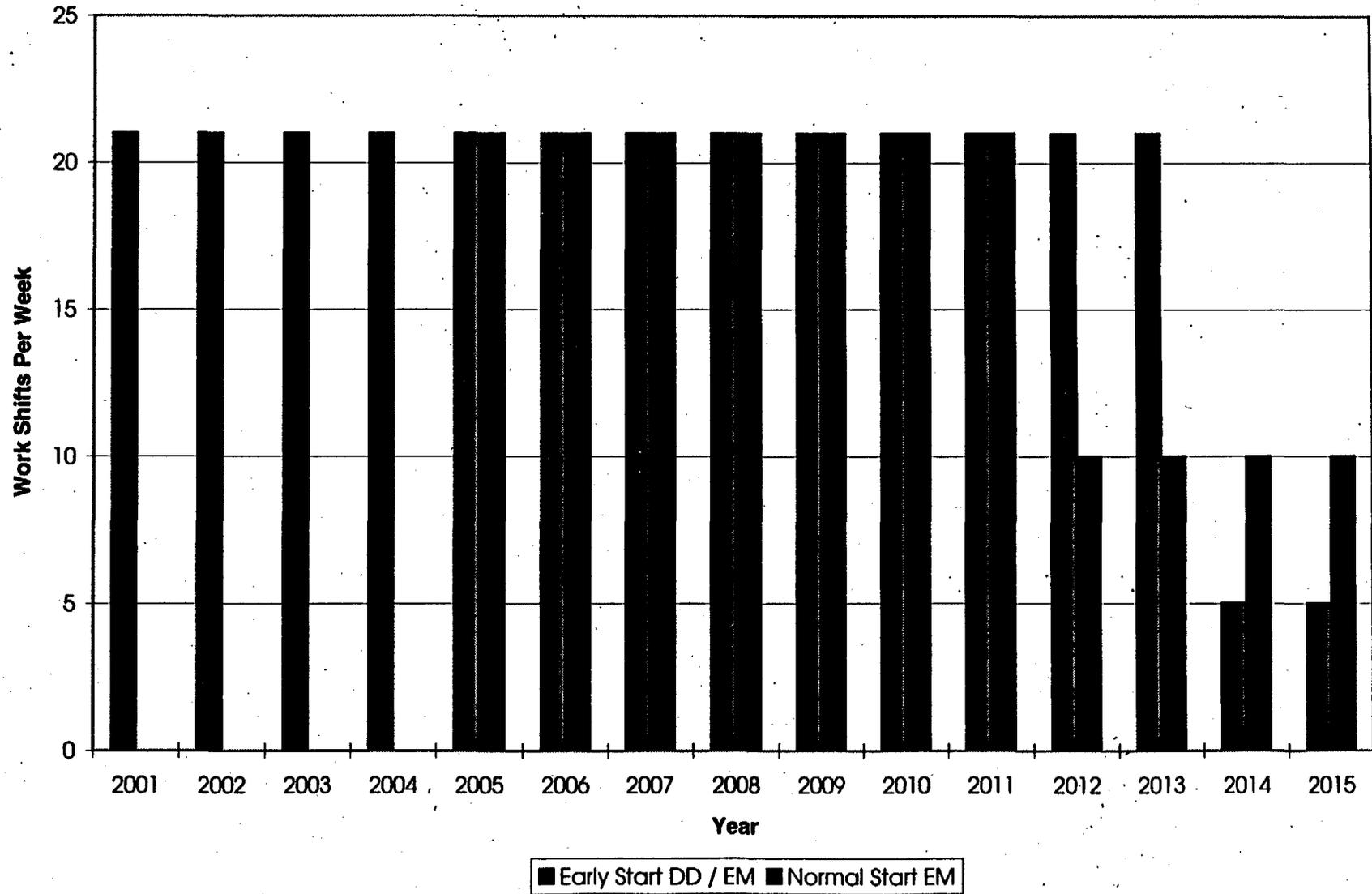


Table C10.3-1 Cost Effect of Early Transfer Facility Implementation (Late Technology)
(Millions of Dollars)

	<u>Expense</u>	<u>Savings</u>
Facility Costs		
Increased Difficulty Costs for Treatment Module	69.30	
Operating Costs		
RBOF, L-Basin Operations 2007-2012		193.20
TF, SF 2001-2005 21 shifts like Co-Disposal	79.50	
TF, SF 2006-2012 21 shifts like GMODS	same	same
TF, SF 2013 21 shifts instead of 10 shifts like GMODS	15.40	
TF, SF 2014-2015 5 shifts instead of 10 shifts like GMODS		14.00
TOTAL	164.20	207.20
	NET	43.00

Table C10.3-2 Cost Effect of Early Transfer Facility Implementation (Early Technology)
(Millions of Dollars)

	<u>Expense</u>	<u>Savings</u>
Facility Costs		
Increased Difficulty Costs for Treatment Module	25.90	
Operating Costs		
RBOF, L-Basin Operations 2007-2008		64.40
TF, SF 2001-2002 21 shifts like Co-Disposal	31.80	
TF, SF 2003-2008 21 shifts like M&D	same	same
TF, SF 2009-2013 10 shifts like M&D	same	same
TF, SF 2014-2015 5 shifts instead of 10 shifts like M&D		14.00
TOTAL	57.70	78.40
	NET	20.70

**Table C10.3-3 Cost Effect of Early Transfer Facility Implementation (EM)
(Millions of Dollars)**

	<u>Expense</u>	<u>Savings</u>
Facility Costs		
New Storage Facility	\$ 47.70	
Increased Difficulty Costs for Treatment Module	68.00	
Operating Costs		
RBOF, L-Basin Operations 2007-2011		161.00
2001-2004	63.60	
21 shifts like Co-Disposal		
TF 2005-2011	same	
21 shifts like EM		same
SF 2005-2011	23.10	
21 shifts like C0-D		
2012-2013	44.20	
10 shifts like EM		
2014-2015		20.00
5 shifts instead of 10 shifts like EM		
TOTAL	198.90	181.00
NET	17.90	

These tables show that the increased facility cost required to maintain flexibility to choose a treatment technology later, and the increased cost to double handle a portion of the SNF, could be offset by the cost saved by closing the basins earlier. Within the assumptions used in the analysis, the cost of these two alternatives are essentially equal. However, for most technologies, a savings is reflected. Use of less conservative assumptions in treatment module add-on costs and consideration of additional factors, such as avoiding the risk of major maintenance expenditures with earlier closure of the existing basins will likely reflect a savings for all technologies. Therefore this option is worthy of further evaluation. The decision date for the treatment module would have to support the design/build schedule shown for that technology (e.g., For Plasma Arc, GMODS, and Dissolve and Vitrify, the design start date is 2001).

Because of the assumptions used in this comparison, great care should be exercised in the use of these savings figures. The assumption about canister re-use will have relatively little impact on the overall system costs. The assumption that the storage facility will remain the same size can be conservative to un-conservative depending on the treatment technology that is ultimately selected. The co-disposal technology generates a larger number of packages to be stored than other technologies. If the follow-on technology is one with an especially small number of packages generated, the size of the storage facility may need to be increased to accommodate storage of the co-disposal packages generated during early years. Table C10.3-4 shows the

number of packages currently planned for several options, and the peak number that would be required if the transfer facility is implemented early.

Table C10.3-4 Maximum Number of Canisters for Early Transfer Facility Implementation

Treatment Technology	Planned Number of Packages	Required Number of Packages for Early Transfer Facility Implementation
Press and Dilute (20%)	400	480
Press and Dilute (2%)	1300	1300
Melt and Dilute	400	480
Plasma Arc	400	860
GMODS	800	880
Dissolve and Vitrify	800	880
Electrometallurgical	0	690

When the increase in required packages, and therefore required storage locations, is approximately 20% (e.g. P&D, M&D, GMODS, and D&V), the increase in the estimate for storage facility construction and operations would increase by less than 5%. This would not change overall conclusions, and would have negligible impact on the overall conclusion. For the Plasma Arc option, the number of required packages, and therefore storage locations, increases by 120% due to early implementation of the transfer facility. This could increase the construction cost of the transfer facility by approximately 25%, and operations costs by 10%. The effect of this change would be an addition of approximately \$14 million to the system cost. Again, this would not change the decision. In summary, early start of the Transfer Facility is an attractive option, with the potential to provide a cost savings, that should be considered further.

10.4 Use the DWPF Glass Waste Storage Building (GWSB) instead of a new interim storage facility

The first GWSB will be filled by 2006 with existing DWPF commitments, and at least one more GWSB will be needed, and more if repository shipments do not begin on schedule. Significant extra GWSB storage space does not appear to be available, and thus, a savings is not apparent. The cost of building additional GWSB storage should be approximately the same as building the interim storage facility. Also, the GWSB is not designed for storage of fissile materials. The Electrometallurgical and Processing options plan on using a nominal amount of space in the GWSB for storage of their glass log waste forms. This has been accounted for in their cost estimates by allocating incremental construction costs for the second GWSB to house the 90 to 120 logs these technologies would generate.

10.5 Do not ship the INEL inventory of aluminum-based SNF to SRS

To assess impact on receipt and transfer facilities, the effect of not receiving the INEL fuel at SRS was evaluated. INEL SNF is not scheduled for shipment until after the peak of the FRR SNF shipments. Therefore, the facility requirements are already determined by the FRR requirements. The only savings possible would be the cost of an additional shift operation for two to three years. Something will have to be done at INEL for this material anyway, so this would not be an overall system savings.

If the "no new facilities" alternative described in Section 10.2 is adopted, deleting the requirement to handle INEL SNF would enable the L-basin pool to be closed three years early, saving approximately \$50 million. Since this alternative costs an additional \$500 million, as compared to the new transfer facility option, this relative savings should not be a deciding factor. There is therefore no economic incentive to avoid shipping INEL aluminum-based SNF to SRS.

10.6 Process the metallic uranium fuel

If metallic uranium fuel is included in Direct Disposal, Co-Disposal or Press and Dilute options, a conditioning module (hot vacuum drying) would have to be added to the transfer facility to stabilize the fuel. Processing this fuel appears to be cost effective. The \$61 million facility and operations cost for a conditioning/HVD module greatly exceeds the estimated \$30 million to process the fuel based on WSRC-RP-95-798.

10.7 Increase the U-235 loading in the co-disposal packages

Current studies have suggested that if depleted uranium were added to the repository disposal package containing the HLW glass logs together with the direct disposal fuel canister, the effective enrichment of the disposal package could be reduced and the U-235 loading increased from 14.4 to 43 kg. This alternative has the potential to reduce the number of packages for the direct disposal options by approximately 30%, resulting in a significant cost savings in canister, treatment, transportation and disposal costs.

11.0 Comparison and Analysis of Results

The comparative cost components for each technology have been shown in the categories of Wet Storage and Handling, Transfer and Packaging, Treatment, Interim Storage, Transportation, Repository Fee, Repository Operations and Adjustments in Tables 4.2-1 and C9.3-2. Some general conclusions can be drawn from inspection of these Tables. Cost differences much less than one hundred million dollars are not significant.

Wet Storage and Handling costs are simply a function of the estimated startup date for the technology. As described earlier these dates are 2001 for direct disposal methods, 2003 for HEU dilution methods, 2006 for advanced treatment options, and 2005 for Uranium separation. The longer a period of time that wet storage and handling are required in the existing basins prior to the startup of the treatment facility, the greater the cost.

Transfer and Packaging costs vary little except that the baseline, reprocessing, enjoys the near term advantage of not requiring the operation of a new facility. Direct disposal and press and dilute are slightly higher due to the need for package venting. Also, technologies with more packages have higher costs.

Treatment and Interim Storage are further broken down into R&D, Facility & Equipment, Operations, and Relative Uncertainty in Tables C11-1 and C11.2.

Treatment costs vary the most. The least expensive technologies in this category are the direct disposal options which have no additional costs for treatment. The treatment costs for press & dilute and melt & dilute are significant and are also assessed moderate relative uncertainty costs. Plasma Arc, GMODS and Electrometallurgical are expected to have significant treatment costs and are subject to high uncertainty. Dissolve & Vitrify and Reprocessing have even higher treatment costs but less uncertainty.

Interim Storage are directly proportional to the number of packages and the length of time to be stored. Electrometallurgical and Reprocessing have the advantage of sharply reducing the SNF to be disposed of by separating and recovering Uranium. Electrometallurgical requires no interim storage in new facilities because the resulting glass logs can be stored in the DWPF with only a small added charge for storage there.

Transportation costs include an assumed fixed cost for cask system development of \$10 million and a cost of \$160 thousand per shipment. The number of shipments for each category is shown in Table C7.3-1.

Repository Fees are assumed to be \$30 million for all technologies, based on a reasonable share of repository development costs.

Repository Operation costs are a function of the number and size of disposal packages for each technology. See Table C7.3-2 for the unit costs.

Adjustments are made only for the cost of a Hot Vacuum Drying facility for those technologies

which maintain Uranium metal and for the value of recovered U-235 for the two technologies capable of recovery. The greater bonus assigned to the Electrometallurgical process versus current Processing reflects the greater recovery expected with the former.

**Table C11.0-1
Treatment Cost Breakdown**

Cost Items	R&D / Development	Facility & Equipment	Operations	D & D	Total
Direct Disposal	0.0	0.0	0.0	0.0	0.0
Co-Disposal	0.0	0.0	0.0	0.0	0.0
Press & Dilute (400)	15.0	31.4	180.3	4.0	230.7
- Treatment Facility (Treatment Module)	15.0	31.4	180.3	4.0	
0		0.0	0.0		
Press & Dilute (1300)	15.0	31.4	180.3	4.0	230.7
- Treatment Facility (Treatment Module)	15.0	31.4	180.3	4.0	
0		0.0	0.0		
Melt and Dilute	20.0	52.9	197.1	4.0	273.9
- Treatment Facility (Treatment Module)	20.0	52.9	197.1	4.0	
0		0.0	0.0		
Plasma Arc	50.0	214.6	184.4	4.0	453.0
- Treatment Facility (Treatment Module)	50.0	214.6	184.4	4.0	
0		0.0	0.0		
GMODS	50.0	169.0	184.4	4.0	407.4
- Treatment Facility (Treatment Module)	50.0	169.0	184.4	4.0	
0		0.0	0.0		
Dissolve and Vitrify	20.0	305.9	387.7	4.0	717.5
- Treatment Facility (Treatment Module)	20.0	305.9	387.7	4.0	
- Adjustment for Relative Uncertainty		0.0	0.0		
Electrometallurgical	40.0	150.5	401.6	4.0	596.1
- Treatment Facility (Treatment Module)	40.0	150.5	401.6	4.0	
0		0.0	0.0		
Processing	0.0	0.0	643.0	0.0	643.0
- Canyon Processing Costs			234.4		
- Canyon De-Inventory Costs			198.0		
- DWPF Glass Log Charges			210.6		

**Table C11.0-2
Interim Dry Storage Breakdown**

Cost Items	R&D / Development	Facility & Equipment	Operations	D & D	Total
Direct Disposal	0.0	56.0	61.8	2.3	120.1
- Vault Storage		56.0	61.8	2.3	
Co-Disposal	0.0	60.1	63.4	2.6	126.0
- Vault Storage		60.1	63.4	2.6	
Press & Dilute (400)	0.0	41.1	54.4	1.4	96.9
- Vault Storage		41.1	54.4	1.4	
Press & Dilute (1300)	0.0	58.5	61.1	2.5	122.1
- Vault Storage		58.5	61.1	2.5	
Melt and Dilute	0.0	41.1	54.4	1.4	96.9
- Vault Storage		41.1	54.4	1.4	
Plasma Arc	0.0	41.1	50.8	1.4	93.3
- Vault Storage		41.1	50.8	1.4	
GMODS	0.0	49.9	53.9	2.0	105.7
- Vault Storage		49.9	53.9	2.0	
Dissolve and Vitrify	0.0	49.9	53.9	2.0	105.7
- Vault Storage		49.9	53.9	2.0	
Electrometallurgical	0.0	3.0	0.0	0.0	3.0
- Vault Storage		0.0	0.0	0.0	
- GWSB Incremental Charges		3.0			
Processing	0.0	20.5	27.7	1.5	49.7
- New Tran & Stor Facility (Trans and Pkg)	0.0	17.5	27.7	1.5	
- GWSB Incremental Charge		3.0			

Appendix D
Direct and Co-Disposal Treatment Technologies

Direct and Co-Disposal Evaluation Team:

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Note: The term "Co-Disposal" as used here refers to the Direct Co-Disposal Option discussed in Volume I of this Report.

Appendix D
Direct and Co-Disposal Treatment Technologies

Table of Contents

- 1.0 Objective**
- 2.0 Approach**
- 3.0 Assumptions**
- 4.0 Direct Disposal in Small Packages**
- 5.0 Co-Disposal with Vitrified Defense High Level Waste**

List of Attachments

Attachments, subjects, along with the lead preparer of each are as follows:

Attachment Subject

- IID-1 Direct Disposal Package Configurations
(Terry Bradley)**
- IID-2 Thermal Evaluation of Direct Disposal Packages
(John McConaghy/Hector Guerra)**
- IID-3 Direct Disposal and Co-Disposal Package Estimate
(Terry Bradley)**
- IID-4 Criticality Analysis for Interim Storage and Transportation
(Kent Parsons)**
- IID-5 Summary of Safeguards and Security Requirements
(Ashok Kapoor)**
- IID-6 Drying & Pressure Buildup Issues for Aluminum-Clad Spent Nuclear Fuel
(Ron Ballinger)**
- IID-7 Recommended Approach to SNF Characterization Requirements
(Ron Ballinger)**

1.0 Objective

This appendix provides supplementary information in support of the Direct Disposal and Co-Disposal overview provided in Vol. I, Sec. 3.3.1.

This supplementary information is formatted consistent with the fact sheets presented at the end of Vol. I, Sec. 3.3. The results of supporting analyses are summarized in this appendix and the analyses included as attachments.

2.0 Approach

The technology required to implement the Direct Disposal Option has been in use for the dry storage of spent nuclear fuel (SNF) at commercial reactors for several years. The objective was to utilize these simple, proven systems and methods to the maximum extent possible.

3.0 Assumptions

The primary assumptions are as follows:

- o The Direct Disposal system will be designed, constructed and operated in accordance with US NRC requirements for interim storage (10 CFR Pt. 72) and transportation (10 CFR Pt. 71).
- o To satisfy repository criticality control requirements, the maximum per-package fissile content will be limited to the values defined in Vol. I, Sec. 3.1.
- o Nominal characterization activities, summarized in Vol. I, Sec. 3.2 and described more fully in Sec. 4.0 below, will be required upon receipt of SNF.

4.0 Direct Disposal in Small Packages

Description

Direct disposal involves packaging and drying the SNF assemblies, placing the assemblies in interim dry storage at SRS, then, upon availability of the repository, transporting the packages to the repository. At the repository, the packages would be placed in a corrosion-resistant disposal overpack immediately before placement into the repository.

The dry storage facility maintains the SNF in a stable, non-degrading condition, and incorporates passive systems with minimal operating and maintenance requirements. By also considering transportation and repository disposal requirements in the design of the interim storage system, the package can be stored in a road-ready condition for later direct

incorporation into the repository. Specially designed casks would be used for on-site transfers and for transportation from SRS to the repository. At the repository, the SNF would not require repackaging, as the entire package would be placed inside the disposal overpack prior to placement into the repository.

Products

A review was conducted of the inventory of SNF which must be accommodated, and for each type of SNF, a proposed package configuration was selected. Att. IID-1 shows the package configurations considered in the study for both the Direct Disposal and Co-Disposal options. The final Direct Disposal package sizes are defined in Table 4.0-1. Inspection of these packages shows that the sizes are shaped by the repository limits on fissile material content (14.4 kg U235 for HEU and 43 kg U235 for LEU) for each disposal package. Also, the small package diameter better accommodates fuels with higher decay heat loading within the 200C peak clad temperature limit (See Att. IID-2) and better accommodates the configuration limits of the existing basins. This results in a larger quantity of smaller packages, as compared to the packages typically used for commercial SNF. Table 4.0-1 summarizes the estimated number of package types and representative loading densities (assemblies per canister for MTR type SNF).

Table 4.0-1 - Quantities and Loading Densities for SNF Packages			
Option	Type A (24"dia. x 10')	Type B 24"dia. x 16')	Type F (24"dia. x 5')
Direct Disposal	80 (32 assy/can)	370 ¹ (64 assy/can)	650 ² (24 assy/can)

¹ Includes 70 Type B packages with 16 assemblies each (long fuel, e.g. NBSR, EBRII, TRR, Slowpoke, KMRR).

² One assembly per package for HFIR, RHF and some other special fuels, due to repository limitations.

Att. IID-3 provides a more detailed description of the package sizes selected by fuel type and a description of the alternate packaging scenarios considered.

The package dimensions selected can be accommodated by all commercially available dry storage and transportation systems. With a vault type storage system, used in combination with a dry transfer facility (packages loaded and sealed remotely), the shield plug, located on the top of each package, could be deleted if similar handling capability is provided at the repository. All available storage systems are expected to be cost competitive.

Secondary Streams

Except for the non-fissile material containing portion of the SNF assemblies, which are cropped, there are no secondary waste streams associated with direct disposal. The assemblies are typically cropped at the sending site to increase transportation efficiency..

Criticality Implications

Repository:

Direct Disposal will result in the placement of highly enriched SNF in the repository, relying on U235 mass content limits, addressed in Vol. I, Sec. 3.1, and spatial distribution to reduce the probability of a criticality. Additional repository considerations are addressed in App. E.

Storage and Transportation:

Direct disposal packages will be provided in accordance with NRC requirements for interim storage (10 CFR Pt. 72) and transportation (10 CFR Pt. 71). Neutron absorbing materials will be required in the internal package structure to meet criticality acceptance limits (Keff less than 0.95 with allowance for bias and uncertainty). Additional details are presented in Att. IID-4.

Proliferation Resistance

The fission products are contained in the direct disposal packages, such that the packages are self-protecting.

A summary of the Safeguards and Security Requirements applicable to the interim storage and transportation of aluminum clad foreign research reactor (FRR) SNF is presented in Att. IID-5.

Technical Maturity

Direct disposal technology is in wide use in commercial SNF dry storage applications. Significant experience has been accumulated with the systems and procedures that would be employed on the FRR SNF. With the direct disposal packages dried and filled with inert gas prior to placement in interim storage, corrosion during interim storage is arrested. The primary area of uncertainty during interim storage is associated with the radiolytic decomposition of bound water and the resulting pressure increase in the package. The following discussions present a) the results of an analysis of this pressure increase, which

concludes that the drying processes used for the drying of commercial SNF is sufficient to maintain pressures during interim storage below direct disposal package limits with considerable margin and b) the recommended approach to characterization requirements..

Drying and Pressure Buildup Issues:

During the drying process all free water will be removed from the fuel. However, bound water will still remain either in the form of hydrated oxides or in the form of chemisorbed water. This water will eventually be released as a result of radiolytic decomposition to form hydrogen and oxygen. The oxygen released will be consumed in the formation of other oxides. However, the hydrogen will remain and will combine with the fill gas to result in an increase in pressure in the storage container/package. This process has been modeled with results summarized in this section and details presented in Att. IID-6. Att. IID-6, Figure 2.1-36 shows a plot of the Free Energy vs. Temperature for hydrated oxides which could be present on the fuel. Att. IID-6, Figure 2.1-35 shows the partial pressure of water over potential oxides as a function of temperature. Based on the Free Energy calculations and typical drying pressures, the only hydrated oxide, which will not be decomposed, is the aluminum oxide mono-hydrate.

Att. IID-6, Figures 3.0-1, -2, -3, & -4 show the evolution of conditions in the package for a typical fuel element loading (32 elements, free volume of 9.4 cubic ft.), and an oxide thickness of 0.0002 in. (0.0005 cm). The ultimate pressure in the package is just under three atmospheres.

Att. IID-6, Figure 3.0-4 shows the terminal pressure in the package as a function of fuel surface oxide thickness for several package free volumes. The oxide thickness ranges modeled bracket the expected fuel conditions.

Based upon this analysis, it is concluded that fuel drying will be adequate to prevent excessive pressure buildup during dry interim storage.

Characterization Requirements:

The approach to characterization requirements used in this study, and which the Team believes are adequate, are summarized in Vol. I, Sec. 3.2. A more detailed discussion of these requirements is presented in Att. IID-7.

Limitations of Direct Disposal

Direct disposal is not suitable for powdered SNF (Sterling Forest Oxide and Mo targets). Also, for direct disposal of the Umetal SNF (EBR II, TRR), a conditioning/HVD Facility must be added, similar to the HVD Facility under development for the Hanford N Reactor

SNF Dry Storage Project. The addition of the facility would add significant complexity and cost to the project for this limited quantity of SNF (~6 cu. m). An alternate approach, to transport the Umetal SNF to Hanford for conditioning and storage at Hanford, is also an undesirable option.

While the technical basis of placing HEU in the repository is well-founded, the licensability of this concept is less certain, as noted in Vol. I, Secs. 2.2 and 3.1, because the regulatory review process has not started for these fuels...

Once the SNF is stored in the direct disposal packages, there is no simple method to sample and characterize the fuel. Thus, adequate characterization must be demonstrated up-front to avoid re-handling.

Facilities and Operations

The existing basins, the Receiving Basin for Off-Site Fuels (RBOF) and L-Basin, will be used for receipt and interim wet storage until a new dry transfer and dry interim storage facilities become operational. Once the dry handling facilities are available, all deliveries will be directed to the new facility, which will be sized to handle the required throughput to receive, package, dry, and loadout to interim storage, as well as, timely reduction of wet storage inventories so that wet storage basins can be decontaminated and decommissioned as soon as possible. The new dry transfer facility will include a module designed for hot vacuum drying of uranium metal SNF if these fuels are included in direct disposal.

These facilities and operations are summarized in Vol. I, Figures 3.2-1, -2 and -3. A more detailed description, including alternatives evaluated, is presented in App. C.

5.0 Co-Disposal with Vitrified Defense High Level Waste

Co-Disposal is similar to Direct Disposal, described in Sec. 4.0 of this appendix, except the package size is varied to take advantage of unused space and the unused U235 mass content capacity in Defense High Level Waste (DHLW) repository waste packages. The Co-Disposal package is ~17" dia. x 10' feet long and loaded to accommodate the repository U235 limits. These packages are loaded, sealed, dried, placed in interim storage at SRS, then shipped to the repository the same as Direct Disposal packages. However, once the Co-Disposal package is received at the repository, it is included in a repository waste package with 5 DHLW glass logs as shown in Vol. I, Figure 3.3-2.

Due to the smaller diameter (17" dia. vs 24" dia.), more waste packages are required for Co-Disposal (1400), as compared to Direct Disposal (1100).

The summaries of products, secondary streams, criticality implications, proliferation resistance and technical maturity information, provided in Sec.4 of this appendix for Direct Disposal, are also applicable for Co-Disposal. However, in respect to criticality implications, the DHLW glass logs within the repository overpack may provide an added benefit in the criticality analysis.

Co-Disposal has the same limitations, as described in Sec. 4 of this appendix, for Direct Disposal. In addition, Co-Disposal has the limitation that the 17" dia. package is very close to the minimum diameter required to store the HFIR and RHF fuels. If a problem is confirmed, options for resolution include use of a package with a slightly larger diameter and, if this is not possible, using a longer repository waste package with short packages, containing the HFIR/RHF assemblies, placed at the ends of the DHLW glass logs, as shown in Att. IID-1, Co-Disposal Concept B (Type F Pkg.). The larger Co-Disposal Concept B waste package would be the same size as currently planned for large diameter multi-purpose canisters (MPC) for commercial SNF.

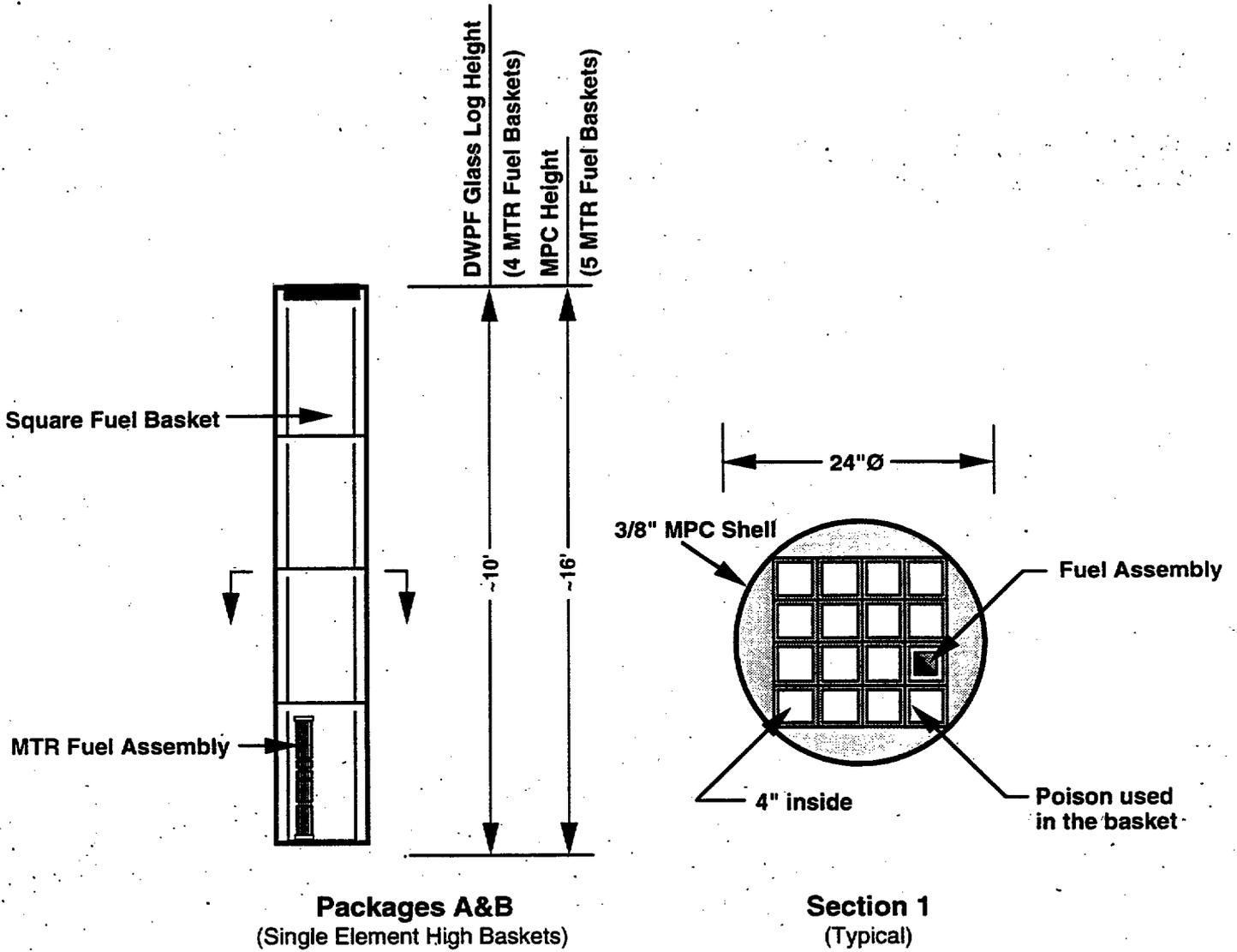
In respect to cost, Co-Disposal has a significant advantage in repository operations cost, due to sharing waste packages with the DHLW. These costs are presented in App. C.

A significant attribute of the Co-Disposal concept, is that it can be used in combination with all the alternative treatment technologies, resulting in significant cost savings..

Attachment IID-1

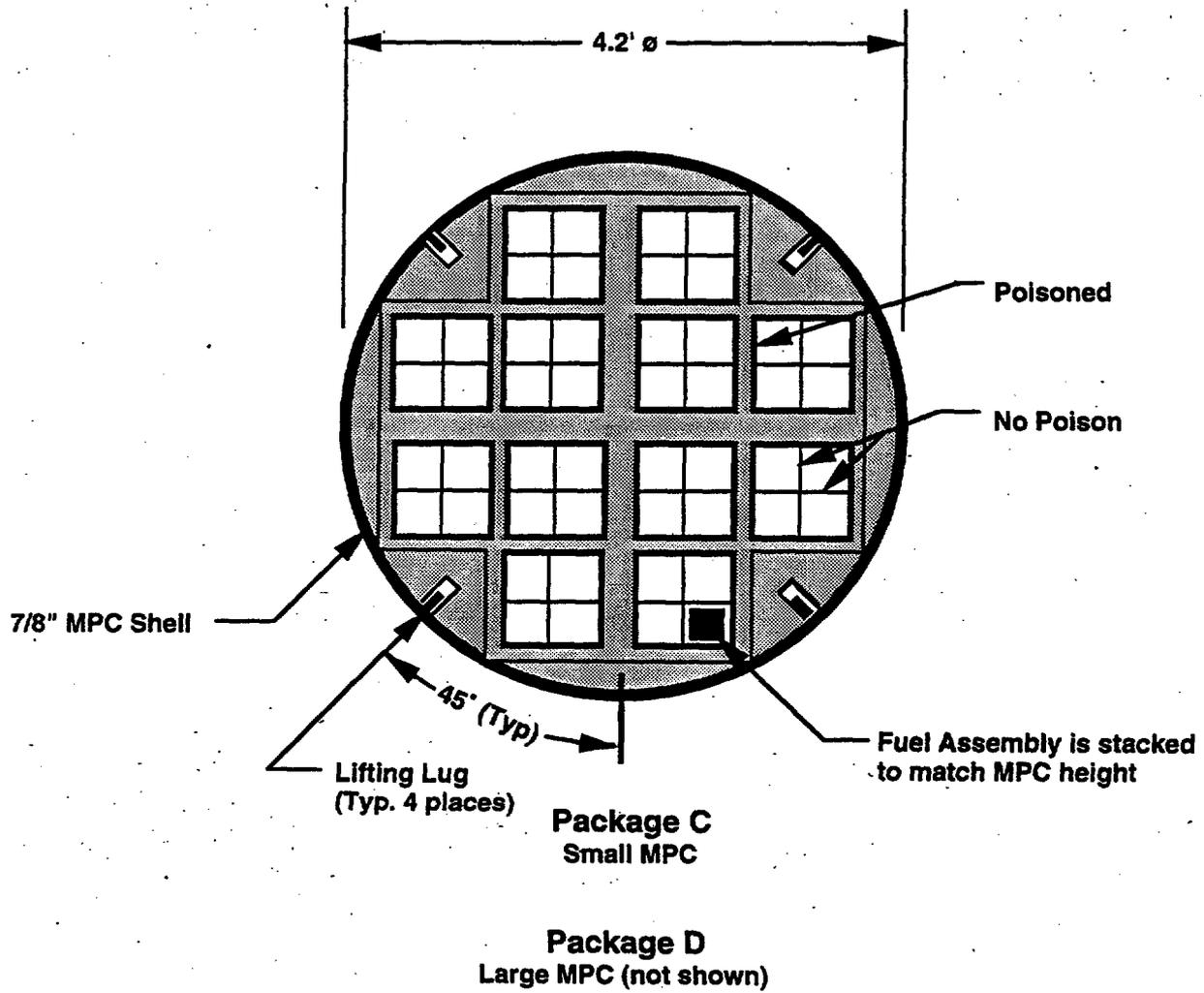
Direct Disposal Package Configurations

**Attachment IID-1
Direct Disposal Package Configurations**

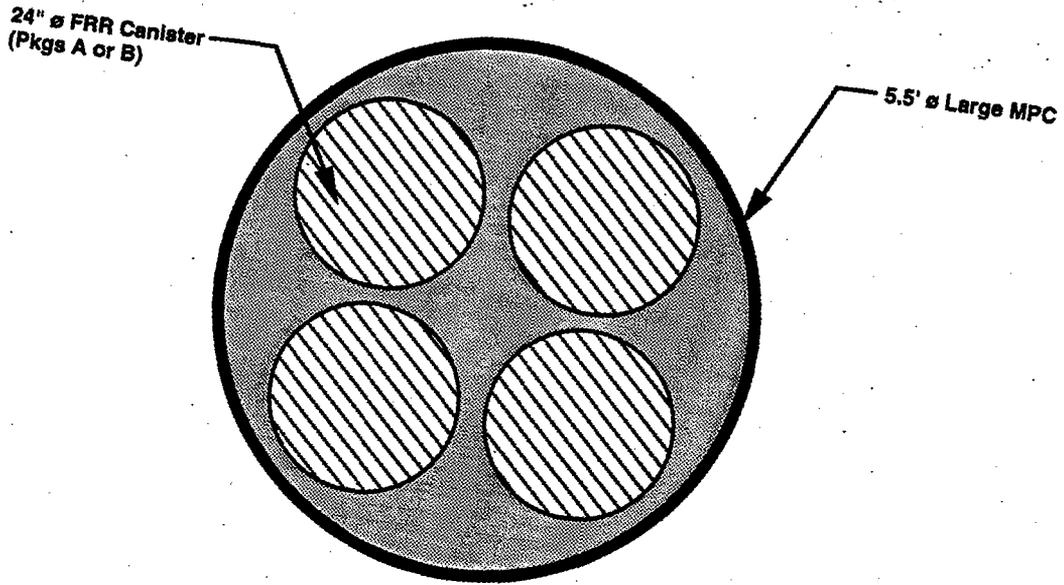


Note: Configuration for MTR fuels shown; baskets may be omitted & basket configuration changed to accommodate other fuels (e.g. NRU).

Attachment IID-1
Direct Disposal Package Configurations (Continued)

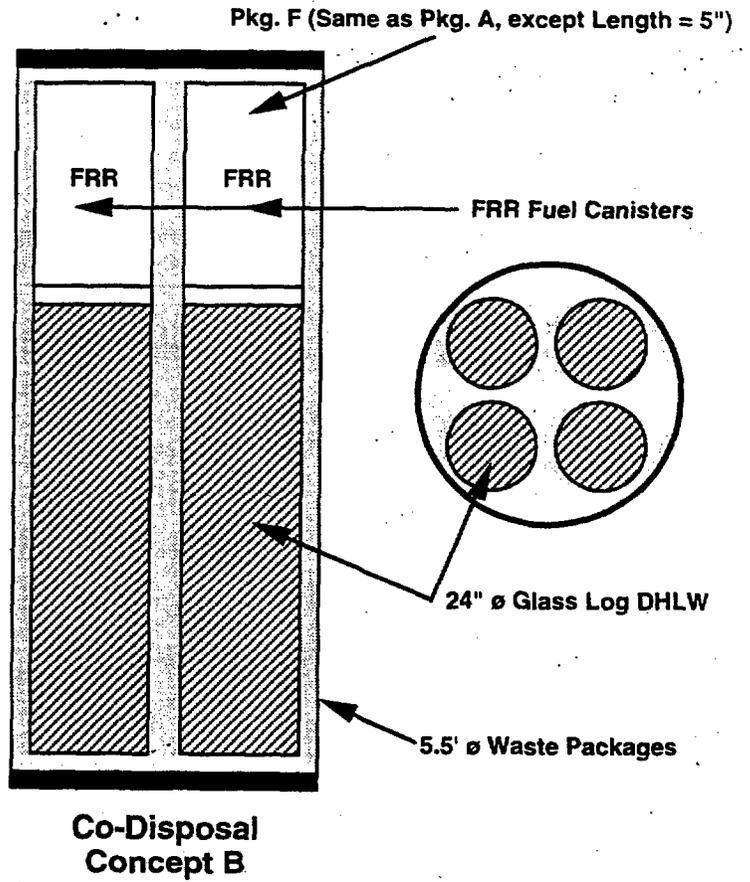
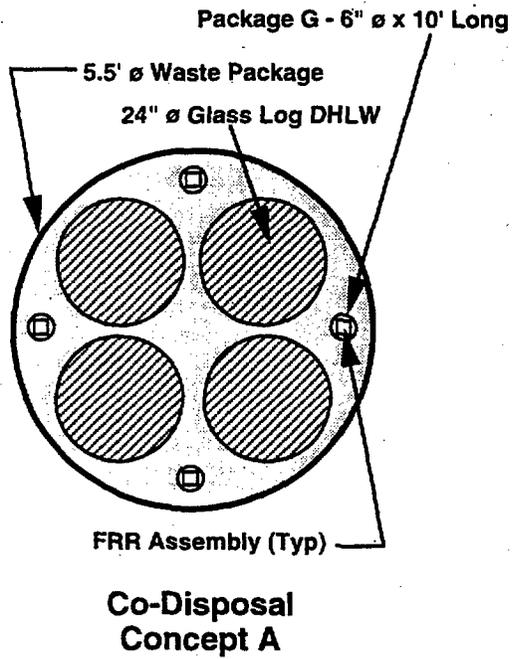


**Attachment IID-1
Direct Disposal Package Configurations (Continued)**

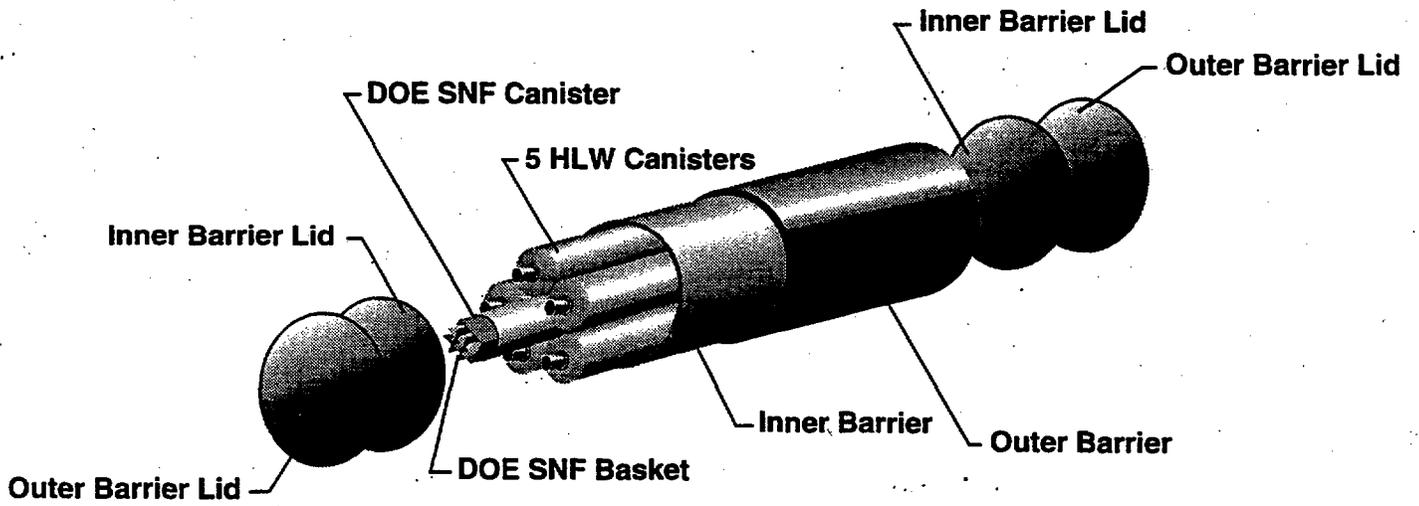


Package E
Storage/Transport Overpack Concept

**Attachment IID-1
Direct Disposal Package Configurations (Continued)**



Attachment IID-1
Direct Disposal Package Configurations (Continued)



Direct Co-Disposal
Final Concept

Attachment IID-2

Thermal Evaluation of Direct Disposal Packages

MEMO

To: Terry Bradley, Ray Conatser
From: John McConaghy
Subject: Thermal Evaluations of RRSNF Canisters
Date: March 18, 1996

I spoke today with Hector Guerra, who works with Dave Muhlbaier at SRS. We had asked Dave whether the proposed configurations for Research Reactor SNF could be engineered to ensure that the peak clad temperature remains less than 200°C. Hector's assessment is shown below.

Based on comparison to previous analyses of dry storage options, five-year old MTRE SNF with typical heat output of 20W would be acceptable for configurations shown in packages A, B, and F by direct comparison. For three-year old MTRE SNF, the heat load increases by a factor of about 3 (63W). This could be engineered for the configurations proposed, but we would not be able to fill all the slots in a given canister. That is, we would have a set of loading rules for canisters with newer SNF.

For HFIR SNF, with significantly higher heat loads, as long as we follow our current concept of one HFIR per canister, he can agree that canisters type A, B, and F can be engineered to dissipate heat rapidly enough to keep the peak clad temperature below 200°C.

The large and small MPC configurations (Packages C and D) are too close to call without extensive analysis. Since the thermal limit for commercial SNF is significantly higher, we cannot conclude that configurations C and D are OK. Since we did not select these MPC configurations in the analysis of different options, this does not affect our outcome.

Attachment IID-3

Direct Disposal and Co-Disposal Package Estimate

Direct Disposal and Co-Disposal Package Estimate

The development of sizes and numbers of packages was an evolving process. The objective was to develop packaging concepts and a rough estimate of the number of packages required. Beginning with the best available information on the many types of fuels, the maximum per-package fissile material content, and facility handling limitations, the first selection of package sizes and numbers was completed and summarized in an estimate, dated 3/6/96. A copy of this estimate, which also presents the assumptions, methodology, and observations, follows this summary.

Subsequent to the 3/6/96 estimate, a combination of revised fuels information and optimization allowed the package counts to be reduced by ~10%.

For Direct Disposal, revised MTR straight-plate type fuel assembly cropped length information (reduced assumed cropped lt. from 32" to 26") permitted the storage of these assemblies 5 high in lieu of 4 high in a Type B (16') Package, resulting in a decrease of 40 Type B Packages. The number of Type A (10') Packages was unchanged, as the number of assemblies per-package was primarily controlled by the fissile material limits. Also, a review of the 3/6/96 estimate indicated the fuel could be packaged more efficiently by mixing\adding MTR assemblies with low Kg U235 loadings to packages containing a single HFIR or RHF assembly, thus, increasing the fissile material content up to the 14.4 Kg limit. This resulted in the reduction of 100 Type F (5') Packages. The number of packages defined in the 3/6/96 estimate and the evolution to the final package estimate is summarized as follows:

<u>Basis\Change</u>	<u>Direct Disposal</u> <u>Package Count</u>			
	<u>Type A</u>	<u>Type B</u>	<u>Type F</u>	<u>Total</u>
3/6/96 Estimate	70	402	745	1217
MTR - 32" to 26" lt.		-40		
Mix HFIR\RHF with MTR			-100	
Final Numbers (Rounded)	80	370	650	1100

For Co-Disposal, the final package configuration (~17' dia. x 10') was identified and selected after the 3/6/96 estimate. The package count was developed considering that a 3x3 assembly array could be stored in the 17" dia. package, as compared to a 4x4 assembly array in the larger 24' dia. Direct Disposal packages. With this 16/9 ratio applied to the Type A and B Direct Disposal Packages, the resulting number of Co-Disposal packages was determined to be 1400.

SRS - FRRTT
Direct Disposal Options
Package/Canister Estimate
Summary/Assumptions
(3\6\96)

This document summarizes the evaluation and assumptions leading to the selection of package sizes for the Direct Disposal in Individual Waste Packages (DD) and Direct Disposal with DWPF LLW Packages - Co-Disposal (C-D) options.

Assumptions are as follows:

- 1) Direct disposal limits of HEU = 14.4 kg/pkg. and LEU = 43 kg/pkg (Ref. Assessment of Technical Issues and Cost for Direct Disposal of FRR, Rev. 0, Dec. 1995, A00000000- 01717-5705-00010).
- 2) All assemblies that are typically cropped (eg MTR fuel) will be cropped.
- 3) The estimate of total packages is intended to be bounding. The MTR fuel is segregated into 8 groups based upon Kg U235/asy.

A summary of attachments and content of each is as follows:

- Att. 1 - Package Sizes Considered
- Att. 2 - Package Summary by Fuel Type
- Att. 3 - Package Selection Details
 - Table - DOM - Domestic (Summary)
 - FRR - FRR (Summary)
 - DETDOM - Domestic (Detailed)
 - FD - Fuel Dims.\Packing Data
- Att. 4 - Packages Selected
- Att. 5 - Alternate Packaging Scenarios Considered

Observations:

- 1) The repository Kg U235 limits drive the DD and C-D packages to the smaller sizes and greater quantities (~1200 packages). As a basis for comparison, the minimum number using package size as the limiting constraint would result in the use of ~160 large diameter MPC type packages.
- 2) Use of only the C-D Concept A (Each C-D package contains 4 Type G packages and 4 DWPF HLW packages) is not a viable standalone concept, as ~ 3000 C-D packages would be required and only ~ 1600 are available.

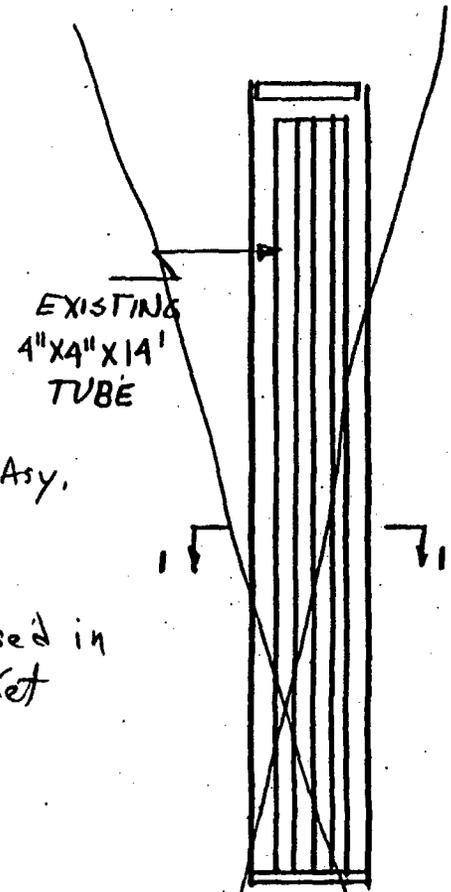
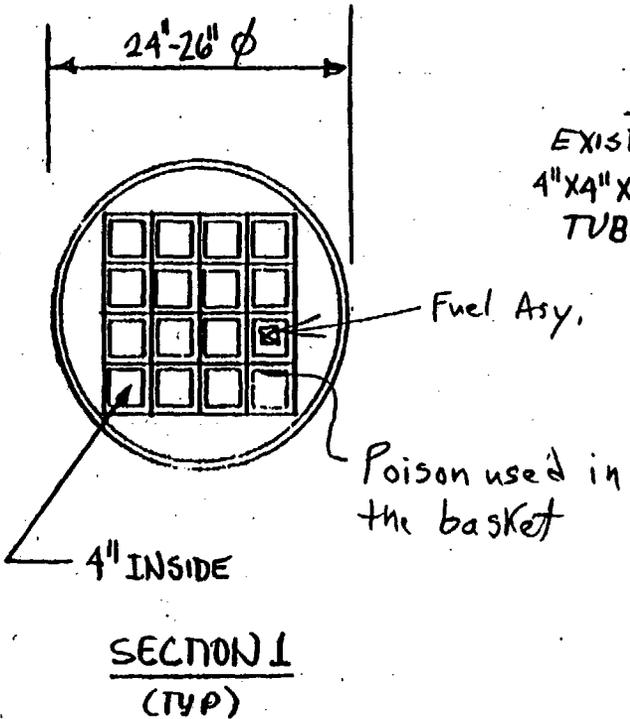
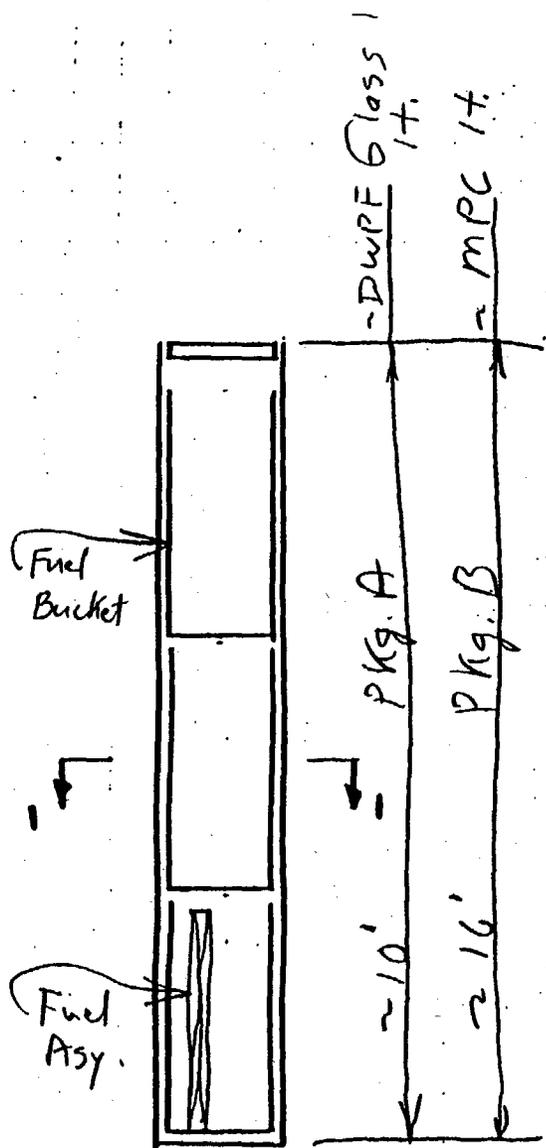
- 3) Co-Disposal using a combination of Concept B Packages (1060 required with each pkg. containing up to 4 Type F pkgs. and 4 DWPF glass logs) and Concept A Packages (280 required with each package containing 4 Type G pkgs. and 4 DWPF glass logs) is a viable option. A total of ~1350 C-D pkgs. are required and ~1600 are available. The Type F pkgs. would be used for the shorter fuels and the Type G pkgs. for the longer fuels.
- 4) The use of small package sizes increases the possibility that the packages can be loaded at the sending site such that the need for repackaging and transfer facilities at SRS can be minimized (egs. Type F pkgs. loaded at the sending site and transported in dual purpose (transportation and storage) casks; Type F pkgs. loaded at the sending site, transported to SRS, then transferred to another storage mode). [Note: The capacity of Type F pkgs. would be cut in half for MTR fuels if the asy.s could not be cropped at the sending site.]

TLB-3\6\96

Atts.

Dry Storage Package Options are attached. The smaller packages will be used for HEU & the larger packages for LEU. Structural materials will be stainless steel A304; neutron absorbing materials will be used in the baskets.

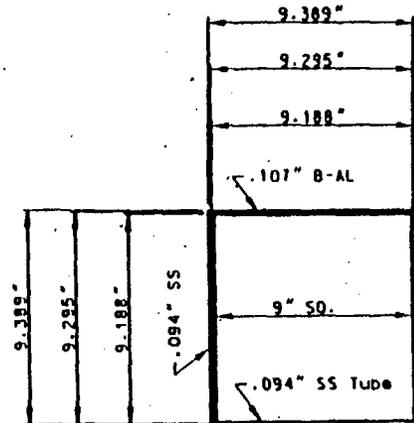
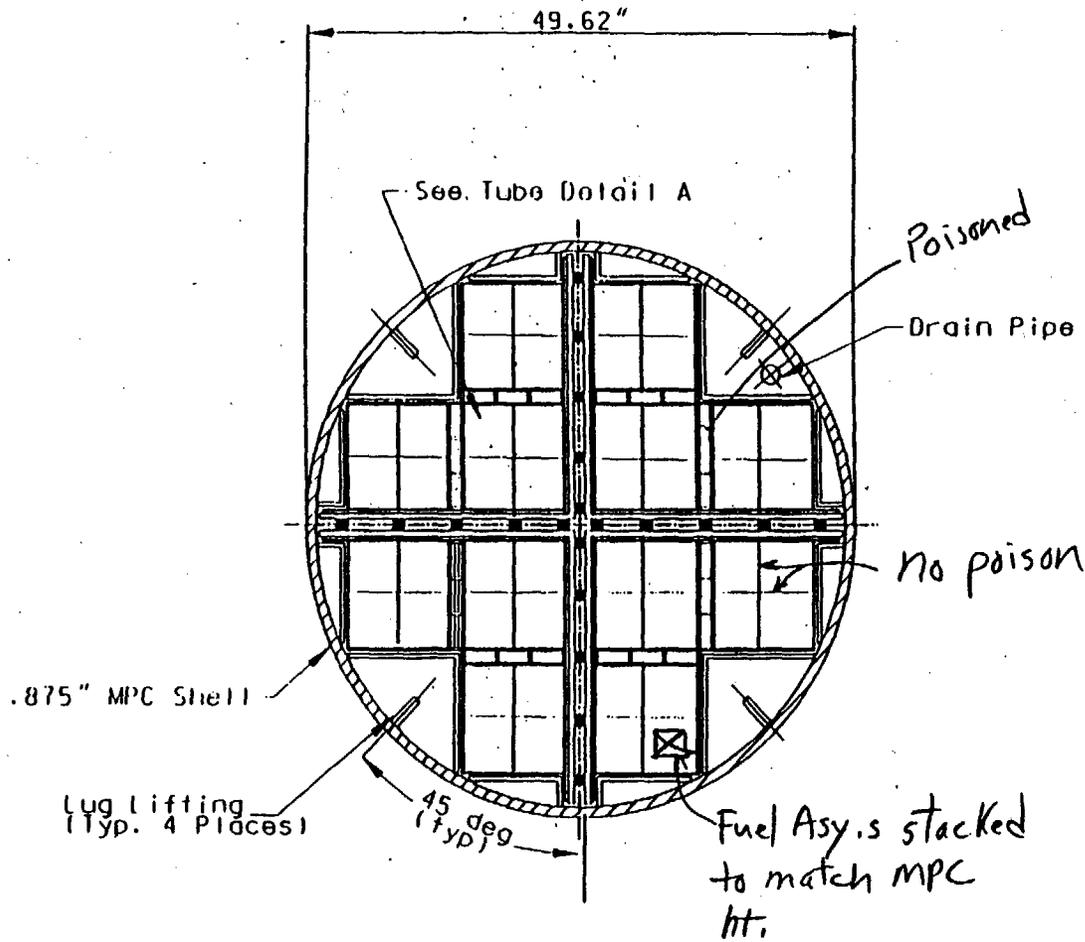
NB 2/15/96



Packages A & B
~~OPTION A~~
 (SINGLE ELEMENT
 HIGH BASKETS)

Note: Configuration for MTR fuels shown; buckets may omitted & basket configuration changed to accommodate other fuels (eg. NRU)

OPTION B
 (THREE ELEMENT
 HIGH TUBES)



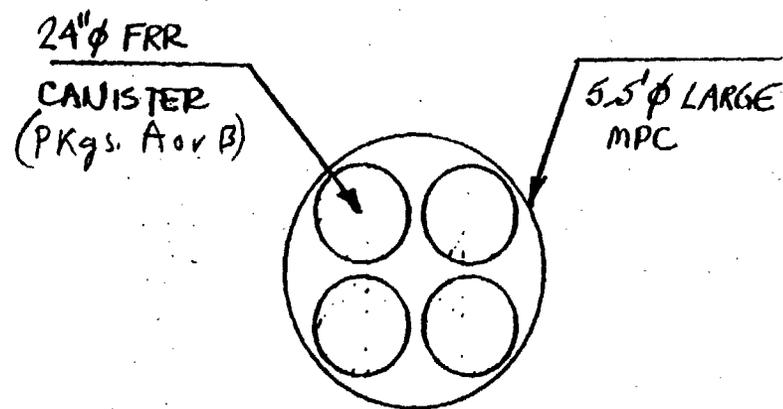
Tube Detail A
(Typical 12 Places)

(Same as pkg. C, except larger dia. & contains 21 PWR cells)

Pkg. C
Small MPC

Pkg. D
Large MPC

Figure 3.1-3 Small 12 PWR MPC (End Section View)

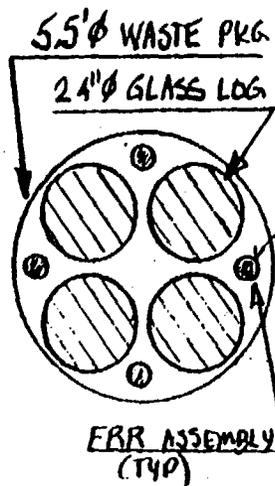


24" ϕ FRR
CANISTER
(PKgs. A or B)

5.5' ϕ LARGE
MPC

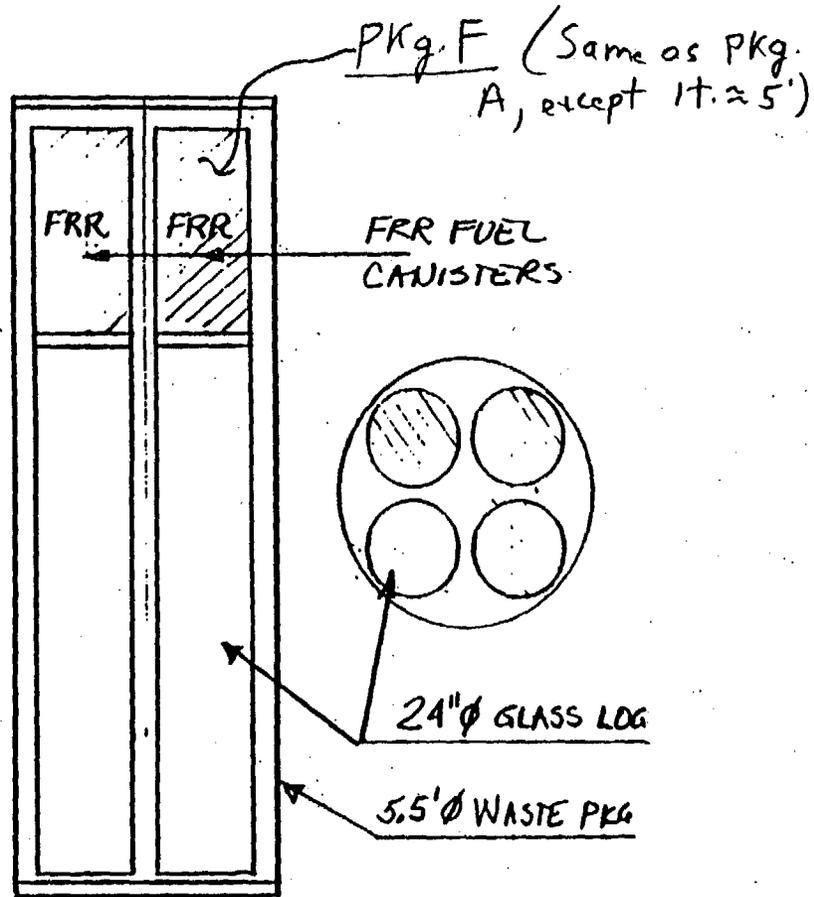
PKg. E

STORAGE / TRANSPORT OVERPACK
CONCEPT



PKg. G - 6" Ø x 10' long

Co-Disposal
PIGGY BACK
CONCEPT A



Co-Disposal
PIGGY BACK
CONCEPT B

Grand Total - Domestic + FRR

Co - Disposal
Concept A Concept B

	<u># Asys.</u>	<u>A Pkgs.</u>	<u>B Pkgs.</u>	<u>F Pkgs.</u>	<u>#C-D Pkgs.</u>	<u>#C-D Pkgs.</u>	<u>#F Pkgs.</u>
HEU-Domestic	15,082	26	157	475	1327	522	1132
-FRR	<u>15,597</u>	<u>35</u>	<u>120</u>	<u>249</u>	<u>935</u>	<u>395</u>	<u>866</u>
	30,679	61	277	724	2262	917	1998
LEU-Domestic	2,112	9	27	21	178	44	108
-FRR	<u>6,137</u>	<u>0</u>	<u>98</u>	<u>—</u>	<u>521</u>	<u>95</u>	<u>380</u>
	8,249	9	125	21	699	139	488
HEU+LEU	38,928	70	402	745	2961	1056	2486

Totals - Domestic

	<u>Co - Disposal</u>	
	<u>Concept A</u>	<u>Concept B</u>

	<u># Asys.</u>	<u>A Pkgs.</u>	<u>B Pkgs.</u>	<u>F Pkgs.</u>	<u>#C-D Pkgs.</u>	<u>#C-D Pkgs.</u>	<u>#F Pkgs.</u>
HEU-MTR	9112	17	129	-	689	159	578
-ATR	3900	-	-	213	371	165	330
-ARMF	85	1	-	-	6	.5	2
-HFBR	220	-	4	-	14	3.5	16
-GENTR	32	1	-	-	2	.5	2
-NBSR	880	-	28	-	220	n/a	n/a
-HFIR	162	-	-	162	n/a	162	162
-TSR	1	-	-	1	1	.25	1
-RHF	4	-	-	2	2	2	2
-Sterl.	200	-	-	11	-	19	19
	asy.bands.						
-Sterl.	226	7	-	-	-	-	-
	<u>pow.can</u>						
	15,082	26	157	475	1,327	522	1,132
LEU-MTR	1849	9	9	21	143	44	108
U Metal - EBRII	120	n/a	14	-	30	n/a	n/a
- TRR	143	n/a	4	-	8	n/a	n/a
	2112	9	27	21	178	44	108
<u>Total - Dom. =</u>	17,194	35	184	496	1,505	566	1,240

Totals - FRR

	<u># Asys.</u>	<u>A Pkgs.</u>	<u>B Pkgs.</u>	<u>F Pkgs.</u>	<u>Co - Disposal</u>		
					<u>Concept A</u>	<u>Concept B</u>	<u>Concept B</u>
					<u>#C-D Pkgs.</u>	<u>#C-D Pkgs.</u>	<u>#F Pkgs.</u>
HEU-MTR	8,682	35	84	68	725	187	577
-Slowpoke	14	-	1	-	2	n/a	n/a
-RHF	181	n/a	n/a	181	n/a	181	181
-Canadian Target	6,720	-	35		210	27	108
LEU-MTR	5,969	-	95		499	95	380
-KMRR	168	-	3		22	n/a	n/a
<u>Total - FRR</u>	21,734	35	218	249	1458	490	1246

Storage & Transport Only (For Comparison Only)
Large Dia. MPC's with Poison - No. kg .U235 Limit

Domestic - 94

FRR - 67

161 - Large Dia. MPC's

or

282 - Small Dia. MPC's

or

~1200 - Dual Purpose Casks (DPC) (32 asy. each)

+ 25 B Pkgs.

(DPC's provided to FRR's such that the fuel
is received at SRS in the DPC's)

The fuel would be repackaged once received at the repository per kg U235 limits, as required.

ATTACHMENT 3
Package Selection Details

D O M
Dry Storage Options
Estimate of Total No. of Pkgs. - Domestic (Summary)
(3/01/96)

Item	ID	U235%	Kg.s\ Asy.	# Asy.s 2006	Direct Dispo.	Direct Dispo.	Direct Dispo.	Direct Dispo.	Co- Dispo.	Co- Dispo.	Co- Dispo.	Co- Dispo.	Stor. & Trans.	Fuel Type
					Max. Asy.s\ Pkg	Pkg. Option	Asys.\ Pkg.	# Pkgs. Req'd	Con. A 4-G's Asy.s\ C-D Pkg.	Con. A 4-G's # C-D Pkgs. Req'd	Con. B 4-F's Asy.s\ C-D Pkg.	Con. B 4-F's # C-D Pkgs. Req'd *	# of D Pkgs. Req'd	
MTR1		HEU	.1-.2 avg. = .15	616	96	B	64	14	12	73	64	14 (56)	3	MTR Group 1
MTR2		HEU	.2-.3 avg. = .25	6551	57	B	57	115	12	547	57	116 (464)	20	MTR Group 2
MTR3		HEU	.3-.4 avg. = .35	45	41	A	41	2	12	4	41	2 (4)	0.00	MTR Group 3
MTR4		HEU	.4-.5 avg. = .45	471	32	F	32	15	12	40	32	15 (30)	2	MTR Group 4
MTR4'		HEU	>.45	1429		F		82		25		12 (24)	5	

D O M
Dry Storage Options
Estimate of Total No. of Pkgs. - Domestic (Summary)
(3/01/96)

					Direct Dispo.	Direct Dispo.	Direct Dispo.	Direct Dispo.	Co-Dispo.	Co-Dispo.	Co-Dispo.	Co-Dispo.	Stor. & Trans.	Fuel Type
Item	ID	U235%	Kg.s\ Asy.	# Asy.s. 2006	Max. Asy.s\ Pkg	Pkg. Option	Asys\ Pkg.	# Pkgs. Req'd	Con. A 4-G's Asy.s\ C-D Pkg.	Con. A 4-F's Asy.s\ C-D Pkg.	Con. B 4-F's Asy.s\ C-D Pkg.	Con. B 4-F's # C-D Pkgs. Req'd*	# of D Pkgs. Req'd	
MTR5		LEU	.1-2 avg. = .15	412	287	B	64	7	12	35	64	7 (28)	2	MTR Group 5
MTR6		LEU	.2-3 avg. = .25	133	172	B	64	2	12	12	64	3 (12)	1	MTR Group 6
MTR7		LEU	.3-4 avg. = .35	2	122	B	64	0.00	12	0.00	64	0.00	1	MTR Group 7
MTR8		LEU	.4-5 avg. = .45	0.00	95	B	64	0.00	12	0.00	64	0.00	0.00	MTR Group 8
MTR8'		LEU	>.45	1300		F		21		96		34 (68)	4	

D O M
Dry Storage Options
Estimate of Total No. of Pkgs. - Domestic (Summary)
(3/01/96)

					Direct Dispo.	Direct Dispo.	Direct Dispo.	Direct Dispo.	Co-Dispo.	Co-Dispo.	Co-Dispo.	Co-Dispo.	Stor. & Trans.	Fuel Type
Item	ID	U235%	Kg.s\ Asy.	# Asy.s 2006	Max. Asy.s\ Pkg	Pkg. Option	Asys.\ Pkg.	# Pkgs. Req'd	Con. A 4-G's Asy.s\ C-D Pkg.	Con. A 4-F's Asy.s\ C-D Pkg.	Con. B 4-F's Asy.s\ C-D Pkg.	Con. B 4-F's # C-D Pkgs. Req'd	# of D Pkgs. Req'd	
7	14	79.45	.6	128	27	F	18	7	8	16	27	4.7 (10)	1	ATR
12	15	82.48	.7	808	20	F	20	41	8	101	20	41 (82)	6.4	ATR
21	16	86.58	.8	2964	18	F	18	165	8	371	18	165 (330)	24	ATR
11	8	90.02	.01	15		A		1		6		.5 (2)	1	ARMF
16	9	93.11	.19	56										
17	10	93.22	.11	2										
18	11	81.10	.12	8										
19	12	91.67	.02	4										
13	102	79.41	.21	220	69	B	69	4	16	14	64	3.5 (16)	1	HFBR
23	97	93.00	.23	32	62	A	32	1			32	.5 (2)		GENTR
24	154	81.02	.11	880	130	B	32	28	4	220	n/a	n/a	5.2	NBSR

DOM
Dry Storage Options
Estimate of Total No. of Pkgs. - Domestic (Summary)
(3/01/96)

					Direct Dispo.	Direct Dispo.	Direct Dispo.	Direct Dispo.	Co-Dispo.	Co-Dispo.	Co-Dispo.	Co-Dispo.	Stor. & Trans.	Fuel Type
Item	ID	U235%	Kg.s\ Asy.	# Asy.s 2006	Max. Asy.s\ Pkg	Pkg. Option	Asys.\ Pkg.	# Pkgs. Req'd	Con. A 4-G's Asy.s\ C-D Pkg.	Con. A 4-F's Asy.s\ C-D Pkg.	Con. B 4-F's Asy.s\ C-D Pkg.	Con. B 4-F's # C-D Pkgs. Req'd	# of D Pkgs. Req'd	
25	103	93.13	9.4	162	1	F	1	162	n/a	n/a	1	162 (162)	8.1	HFIR
28	270	93.48	8.6	1	1	F	1	1			1	.25 (1)		TSR
29	38	86.96	1.1	19	13									CX
30	211	96.92	12.2	18	1									SPR III
31	212	96.77	30.0	1	0									SPR I, II, III
47	179	81.44	5.2	4	2	F	2	2			2	2 (2)		RHF
51	226	84.96	.73	200	19	F	19	11	19	11	19	19 (19)	2	Sterling

DOM
Dry Storage Options
Estimate of Total No. of Pkgs. - Domestic (Summary)
(301196)

					Direct Dispo.	Direct Dispo.	Direct Dispo.	Direct Dispo.	Co-Dispo.	Co-Dispo.	Co-Dispo.	Co-Dispo.	Stor. & Trans.	Fuel Type
Item	ID	U235%	Kg.s\ Asy.	# Asy.s 2006	Max. Asy.s\ Pkg.	Pkg. Option	Asys.\ Pkg.	# Pkgs. Req'd	Con. A 4-G's Asy.s\ C-D Pkg.	Con. A 4-F's Asy.s\ C-D Pkg.	Con. B 4-F's Asy.s\ C-D Pkg.	Con. B 4-F's # C-D Pkgs. Req'd	# of D Pkgs. Req'd	
52	225	92.69	.45	226	32	A	32	7	4	57	32			Sterling
37	59	.25	2.0	120 cans	21	B	9	14	4	n/a	n/a		6	EBR II U metal
53	269	.63	1.0	143	43	B	43	4	20	n/a	n/a		1	TRR U metal

FRR
Dry Storage Options
Estimate of Total No. of Pkgs. - FRR
(3/01/96)

Item	ID	U235 %	Kg.s/Asy.	# Asy.s 2006	Direct Dispo. Max. Asy.s/ Pkg	Direct Dispo. Pkg. Option	Direct Dispo. Asys. Pkg.	Direct Dispo. # Pkgs. Req'd	Co-Dispo. Con. A 4-G's Asy.s/ C-D Pkg.	Co-Dispo. Con. A 4-G's # C-D Pkgs. Req'd	Co-Dispo. Con. B 4-F's Asy.s/ C-D Pkg.	Co-Dispo. Con. B 4-F's # C-D Pkgs. Req'd *	Stor. & Trans. # of D Pkgs. Req'd	Fuel Type
MTR1		HEU	.1-2 avg. = .15	3415	96	B	64	54	12	285	64	54 (216)	11	MTR Group 1
MTR2		HEU	.2-3 avg. = .25	1684	57	B	57	30	12	141	57	30 (120)	5	MTR Group 2
MTR3		HEU	.3-4 avg. = .35	1417	41	A	41	35	12	118	41	35 (105)	5	MTR Group 3
MTR4		HEU	.4-5 avg. = .45	2166	32	F	32	68	12	181	32	68 (136)	7	MTR Group 4
MTR5		LEU	.1-2 avg. = .15	1312	287	B	64	21	12	110	64	21 (84)	4	MTR Group 5

FRR
Dry Storage Options
Estimate of Total No. of Pkgs. - FRR
(3/01/96)

Item	ID	U235 %	Kg.s\ Asy.	# Asy.s. 2006	Direct Dispo. Max. Asy.s\ Pkg	Direct Dispo. Pkg. Option	Direct Dispo. Asys.\ Pkg.	Direct Dispo. # Pkgs. Req'd	Co-Dispo. Con. A 4-G's Asy.s\ C-D Pkg.	Co-Dispo. Con. A 4-G's # C-D Pkgs. Req'd	Co-Dispo. Con. B 4-F's Asy.s\ C-D Pkg.	Co-Dispo. Con. B 4-F's # C-D Pkg.s Req'd *	Stor. & Trans. # of D Pkgs. Req'd	Fuel Type
MTR6		LEU	.2-.3 avg. = .25	1186	172	B	64	19	12	99	64	19 (76)	4	MTR Group 6
MTR7		LEU	.3-.4 avg. = .35	1118	122	B	64	18	12	94	64	18 (72)	4	MTR Group 7
MTR8		LEU	.4-.5 avg. = .45	2353	95	B	64	37	12	196	64	37 (148)	7	MTR Group 8
Slow poke		HEU	.44	14	32	B	14	1				n/a		
RIIF		HEU	8.5	181	1	F	1	181	n/a	n/a	1	181 (181)	12	
KMRR		HEU	.1-.25	168	172	B	64	3	8	22	n/a	n/a	1	
Canad. Target		HEU	.04	6720	360	B	192	35	32	210	256	27 (108)	7	

DETDOM
Dry Storage Options
Estimate of Total No. of Pkgs. - Domestic (Detailed)
(3/01/96)

Item	ID	UJ235 %	Kg.s\ Asy.	# Asy.s 2035	Direct Dispo. Max. Asy.s\ Pkg	Direct Dispo. Pkg. Option	Direct Dispo. Asys.\ Pkg.	Direct Dispo. # Pkgs. Req'd	Co-Dispo. Con. A 4-G's Asy.s\ C-D Pkg.	Co-Dispo. Con. A 4-G's # C-D Pkgs. Req'd	Co-Dispo. Con. B 4-F's Asy.s\ C-D Pkg.	Co-Dispo. Con. B 4-F's # C-D Pkgs. Req'd	Stor. & Trans. # of D Pkgs. Req'd	Fuel Type [MTR Grp.]
2	104	76.56	.2	6537	72	B	72	91	16	409	64	103	16	MTR [2]
3	21	83.84	.08	68	72+	B	72	1	16	5	64	1	n/a	MTR [1]
7	14	79.45	.6	128	27	A	18	7.1	8	16	27	4.7	n/a	ATR
8	101	77.55	.19	20	72	A	20	.3	16	2	20	.5	n/a	MTR [1]
9	141	88.53	.6	24	27	A	24	.5	12	2.1	24	.5	n/a	MTR [4+]
10	161	79.87	.15	17	72+	A	17	.3	16	1.1	17	.4	n/a	MTR [1]
11	8	90.02	.01	15	72+	A	15	.3	16	1	15	.3	n/a	armf
12	15	82.48	.7	808	20	A	20	41	8	101	20	.6	6.4	ATR
13	102	79.41	.21	220	69	B	69	3.2	16	14	64	3.5	6	HEBR

DETDOM
Dry Storage Options
Estimate of Total No. of Pkgs. - Domestic (Detailed)
(3/01/96)

					Direct Dispo.	Direct Dispo.	Direct Dispo.	Direct Dispo.	Co-Dispo.	Co-Dispo.	Co-Dispo.	Co-Dispo.	Stor. & Trans.	Fuel Type
Item	ID	U235 %	Kg.s\ Asy.	# Asy.s 2035	Max. Asy.s\ Pkg	Pkg. Option	Asys.\ Pkg.	# Pkgs. Req'd	Con. A 4-G's Asy.s\ C-D Pkg.	Con. A 4-G's # C-D Pkgs. Req'd	Con. B 4-F's Asy.s\ C-D Pkg.	Con. B 4-F's # C-D Pkg.s Req'd	# of D Pkgs. Req'd	[MTR Grp.]
14	142	87.47	.59	32	27	A	27	1.2	16	2	27	1.2	n/a	MTR plate [4+]
15	281	93.07	.14	26	72+	A	26	1			26	.5		MTR [1]
16	9	93.11	.19	56	72+	A	56	1			56	1		armf
17	10	93.22	.11	2	72+	A	15	1			15	.25		armf
18	11	81.10	.12	8	72+									armf
19	12	91.67	.02	4	72+									armf
20	35	0.64	1.4	1	31									Core filter
21	16	86.58	.8	2964	18	F	18	165	8	371	18	165	24	ATR
22	159	85.73	.15	86	72+	B	80	2			64	1.5		MTR [1]

DETDOM
Dry Storage Options
Estimate of Total No. of Pkgs. - Domestic (Detailed)
(3/01/96)

					Direct Dispo.	Direct Dispo.	Direct Dispo.	Direct Dispo.	Co-Dispo.	Co-Dispo.	Co-Dispo.	Co-Dispo.	Stor. & Trans.	Fuel Type
Item	ID	U235 %	Kg.s\ Asy.	# Asy.s 2035	Max. Asy.s\ Pkg	Pkg. Option	Asys.\ Pkg.	# Pkgs. Req'd	Con. A 4-G's Asy.s\ C-D Pkg.	Con. A 4-G's # C-D Pkgs. Req'd	Con. B 4-F's Asy.s\ C-D Pkg.	Con. B 4-F's # C-D Pkgs. Req'd	# of D Pkgs. Req'd	[MTR Grp.]
23	97	93.00	.23	32	62	A	32	1			32	.5		gentr
24	154	81.02	.11	880	130	B	32	28	4	220	n/a	n/a	5.2	nbsr
25	103	93.13	9.4	162	1	F	1	162	n/a	n/a	1	162	8.1	hfir
26	31	85.71	.15	41	96	B	64	2			41	.75		MTR plate [1]
27	165	19.82	.32	32	134	A	32	1			32	.5		MTR plate [3]
28	270	93.48	8.6	1	1	F	1	1			1	.25		TSR
29	38	86.96	1.1	19	13									CX
30	211	96.92	12.2	18	1									SPR III

DETDOM
Dry Storage Options
Estimate of Total No. of Pkgs. - Domestic (Detailed)
(3/01/96)

					Direct Dispo.	Direct Dispo.	Direct Dispo.	Direct Dispo.	Co-Dispo.	Co-Dispo.	Co-Dispo.	Co-Dispo.	Stor. & Trans.	Fuel Type
Item	ID	U235 %	Kg.s\ Asy.	# Asy:s 2035	Max. Asy.s\ Pkg	Pkg. Option	Asys.\ Pkg.	# Pkgs. Req'd	Con. A 4-G's Asy.s\ C-D Pkg.	Con. A 4-G's # C-D Pkgs. Req'd	Con. B 4-F's Asy.s\ C-D Pkg.	Con. B 4-F's # C-D Pkg.s Req'd	# of D Pkgs. Req'd	[MTR Grp.]
31	212	96.77	30.0	1	0									SPR I,II,III
35	5	92.94	.65	4	22	F	8	1		1	8	.25		MTR plate? [4+]
36	17	93.17	.75	4	19									MTR plate? [4+]
37	59	.25	2.0	120	21	B	21	6				n/a		EBR II - umetal
38	78	73.12	.1	153	144	B	64	3			64	3		MTR plate? [1]

DETDOM
Dry Storage Options
Estimate of Total No. of Pkgs. - Domestic (Detailed)
(3/01/06)

Item	ID	U235 %	Kg.s\ Asy.	# Asy.s 2035	Direct Dispo. Max. Asy.s\ Pkg	Direct Dispo. Pkg. Option	Direct Dispo. Asys.\ Pkg.	Direct Dispo. # Pkgs. Req'd	Co-Dispo. Con. A 4-G's Asy.s\ C-D Pkg.	Co-Dispo. Con. A 4-G's # C-D Pkgs. Req'd	Co-Dispo. Con. B 4-F's Asy.s\ C-D Pkg.	Co-Dispo. Con. B 4-F's # C-D Pkgs. Req'd	Stor. & Trans. # of D Pkgs. Req'd	Fuel Type [MTR Grp.]
39	123	88.43	1.0	15	14	F	14	1		2	14	1.25		MTR plate [4+]
40	136	81.06	.81	16	17	F	16	1		2	16	.25		MTR plate [4+]
41	143	87.26	1.2	52	12	F	12	5	12	5	12	5		MTR [4+]
42	155	19.81	.88	8	48	F	8	1		1	8	.25		MTR [4+]
43	157	93.66	.8	4	18	F								MTR [4+]
44	163	82.21	1.0	17	14	F	14	2			2	14	2	MTR [4+]
45	164	16.03	1.0	14	43	F	14	1			4	16	.25	MTR [4+]

DETDOM
Dry Storage Options
Estimate of Total No. of Pkgs. - Domestic (Detailed)
(3/01/96)

Item	ID	U235 %	Kg.s\ Asy.	# Asy.s 2035	Direct Dispo. Max. Asy.s\ Pkg	Direct Dispo. Pkg. Option	Direct Dispo. Asys.\ Pkg.	Direct Dispo. # Pkgs. Req'd	Co- Dispo. Con. A 4-G's Asy.s\ C-D Pkg.	Co- Dispo. Con. A 4-G's # C-D Pkgs. Req'd	Co- Dispo. Con. B 4-F's Asy.s\ C-D Pkg.	Co- Dispo. Con. B 4-F's # C-D Pkgs. Req'd	Stor. & Trans. # of D Pkgs. Req'd	Fuel Type [MTR Grp.]
46	162	11.01	.3	2	143									MTR [7]
47	179	81.44	5.2	4	2	F	2	2			2	2		RIIF
48	180	90.13	.59	13	27	F	13	1			1	13	.25	MTR [4+]
51	226	84.96	.73	34	19									sterl.
52	225	92.69	.45	226	32									sterl.
53	269	.63	1.0	143	43	B	43	4				n/a		TRR umetal
54	276	14.67	.62	8	69	F	8	1			8	.25		MTR [8+]
55	279	88.48	.75	8	19	F	8	1		2	8	.25		MTR [4+]

DETDOM
Dry Storage Options
Estimate of Total No. of Pkgs. - Domestic (Detailed)
(3/01/96)

					Direct Dispo.	Direct Dispo.	Direct Dispo.	Direct Dispo.	Co-Dispo.	Co-Dispo.	Co-Dispo.	Co-Dispo.	Stor. & Trans.	Fuel Type
Item	ID	U235 %	Kg.s\ Asy.	# Asy.s 2035	Max. Asy.s\ Pkg	Pkg. Option	Asys.\ Pkg.	# Pkgs. Req'd	Con. A 4-G's Asy.s\ C-D Pkg.	Con. A 4-G's # C-D Pkgs. Req'd	Con. B 4-F's Asy.s\ C-D Pkg.	Con. B 4-F's # C-D Pkgs. Req'd	# of D Pkgs. Req'd	[MTR Grp.]
61	87	90.09	.17	27	84	A	27	1			27	.5		MTR [1]
62	88	19.79	.22	133	195	B	80	1.7	16	9	64	3	.4	MTR [6]
63	121	95.12	.28	14	57	A	14	1			14	.25		MTR [2]
64	122	19.8	.31	13	138	A	13	1			13	.25		MTR [3]
65	135	88.56	.4	471	36	A	36	13	12	40	36	13	1.4	MTR plate [4]
66	158	19.67	.01	414	4300?	B	64	6.5	12	35	64	7	1.3	MTR plate [5]

DETDOM
Dry Storage Options
Estimate of Total No. of Pkgs. - Domestic (Detailed)
(3/01/96)

					Direct Dispo.	Direct Dispo.	Direct Dispo.	Direct Dispo.	Co-Dispo.	Co-Dispo.	Co-Dispo.	Co-Dispo.	Stor. & Trans.	Fuel Type
Item	ID	U235 %	Kg.s\ Asy.	# Asy.s 2035	Max. Asy.s\ Pkg	Pkg. Option	Asys.\ Pkg.	# Pkgs. Req'd	Con. A 4-G's Asy.s\ C-D Pkg.	Con. A 4-G's # C-D Pkgs. Req'd	Con. B 4-F's Asy.s\ C-D Pkg.	Con. B 4-F's # C-D Pkgs. Req'd	# of D Pkgs. Req'd	[MTR Grp.]
67	177	91.13	.02	124	7207	B	64	2	12	11	64	3	.3	MTR plate [1]
68	178	20.00	1.13	22	38	A	22	1	12	3	22	.5		MTR plate [8+]
69	181	19.28	1.4	178	30	F	30	6	12	15	30	6	.6	MTR plate [8+]
70	274	93.03	.17	26	84	A	26	1			26	.5		MTR plate [1]
71	275	19.73	1.34	28	32	F	32	1	12	3	28	1	n/a	MTR plate [8+]

DETDOM
Dry Storage Options
Estimate of Total No. of Pkgs. - Domestic (Detailed)
(3/01/96)

Item	ID	U235 %	Kg.s\ Asy.	# Asy.s 2035	Direct Dispo. Max. Asy.s\ Pkg	Direct Dispo. Pkg. Option	Direct Dispo. Asys.\ Pkg.	Direct Dispo. # Pkgs. Req'd	Co-Dispo. Con. A 4-G's Asy.s\ C-D Pkg.	Co-Dispo. Con. A 4-G's # C-D Pkgs. Req'd	Co-Dispo. Con. B 4-F's Asy.s\ C-D Pkg.	Co-Dispo. Con. B 4-F's # C-D Pkgs. Req'd	Stor. & Trans. # of D Pkgs. Req'd	Fuel Type [MTR Grp.]
72	272	92.45	.01	256	1440?	B	64	4	12	22	64	4	8	MTR plate [8+]
73	273	19.72	1.12	25	38	A	25	1	12	2	25	1	n/a	MTR plate [8+]
74	278	19.81	1.07	552	40	A	40	1	16	35	40	14	1.4	MTR [8+]
75	277	16.85	.75	131	57	A	57	2.3	16	9	57	3	.4	MTR [8+]
76	144	90.40	.75	1218	19	F	19	64	16	76	19	64	3	MTR 4+]
77	145	89.58	.17	28	84	A	28	1	16	2	28	1	n/a	MTR [1]
78	146	19.70	.94	28	45	A	28	1	16	2	28	1	n/a	MTR [8+]

DETDOM
Dry Storage Options
Estimate of Total No. of Pkgs. - Domestic (Detailed)
(3/01/96)

					Direct Dispo.	Direct Dispo.	Direct Dispo.	Direct Dispo.	Co-Dispo.	Co-Dispo.	Co-Dispo.	Co-Dispo.	Stor. & Trans.	Fuel Type
Item	ID	U235 %	Kg.s\ Asy.	# Asy.s 2035	Max. Asy.s\ Pkg	Pkg. Option	Asys.\ Pkg.	# Pkgs. Req'd	Con. A 4-G's Asy.s\ C-D Pkg.	Con. A 4-G's # C-D Pkgs. Req'd	Con. B 4-F's Asy.s\ C-D Pkg.	Con. B 4-F's # C-D Pkg.s Req'd	# of D Pkgs. Req'd	[MTR Grp.]
79	280	19.54	1.24	45	34	A	45	1	16	2	34	1.4	n/a	MTR [8+]
80	287	19.71	.84	27	51	A	27	1	16	2	27	1	n/a	MTR [8+]

F D
Dry Storage Options
Fuels Dimensional Packaging Data
 (3\01\96)

Fuel	Cropped Dims. (ins.)	# per 16' high stack	# per 10' high stack	# per 5'-4" high stack	24" dia. pkg.-array-asy. cap.	Small MPC asy. cap.	Large MPC asy. cap.		
MTR-Plate	3.25x3.25x26	5	4	2	4x4=16	240	420		
MTR-Curve Plate	3x3x26	5	4	2	4x4=16	240	420		
MTR-Tube	4" dia. x26	5	4	1 (possibly 2)	4x4=16	240	420		
NRUNRX	1.9" dia. (max.) x10.5' (max.)	1	n/a	n/a	4x4=16 cells w\ 4\cell = 64	192	336	Criticality ok?	
RIIF	16" dia. x16" high	4	n/a	n/a	1 asy.	4	16		
TRIGA	1.5" dia. x30	5	3	1	4x4=16 cells w\ 5\cell = 80	300	525	Criticality ok?	
ATR	4.25 max. dim. x50	3	2	1	3x3=9	72 (2PWRcell)	126		
ARMF/ CFRMF	3.25x3.25x26	5	4	1 (possibly 2)	4x4=16	240	420		

F D
Dry Storage Options
Fuels Dimensional Packaging Data
(3\01\96)

HFBR (MTR)	3.25x3.25x 26	5	4	1 (possibly 2)	4x4=16	240	420		
Fuel	Cropped Dims. (ins.)	# per 16' high stack	# per 10' high stack	# per 5'-4" high stack	24" dia. pkg.-array- asy. cap.	Small MPC asy. cap.	Large MPC asy. cap.		
Id 35	9"dia.x30							Core filter	
GENTR	2.75"dia.x17							Stacked disks	
NBSR	3.5x3.5x69 (lt.?)	2	1	n/a	4x4=16	96	168	17 curved plates	
HFIR	17.5"dia.x32	4	3	1	1 asy.	16	20		
TSR	24"spherical annulus								
CX	n/a								
SPR I,II,III	n/a								
Sterling	powder can bundles - 3.5"x3.5"x10'	1	0	0	4x4x16	48	84		
	Asy. bundles - 3"x3"x26"	5	4	2	4x4x16	240	420		

F D
Dry Storage Options
Fuels Dimensional Packaging Data
(3\01\96)

TRR U metal	1"dia.x10'	1	n/a	n/a	4x4=16 cells w\ 5'cell=80	240	420		
EBR-II U metal	5"dia.x14' can	1	n/a	n/a	3x3=9	BWRMPC-24 PWR-12	BWRMPC-24 PWR-21		
Fuel	Cropped Dims. (ins.)	# per 16' high stack	# per 10' high stack	# per 5'-4" high stack	24" dia. pkg.-array- asy. cap.	Small MPC asy. cap.	Large MPC asy. cap.		
Slowpoke	2"dia.x 9.6' can (contains 150-160 pins)								
Canadian Targets	2.75"dia. x11"	12	8	4	4x4=16	576	1008		
Korean (KMRR)	same as NRU								

Baseline

Direct Disposal - Pkg. A = 70
Pkg. B = 360
Pkg. F = 750
1180

If handling limits cause problems most of the B's can be converted to A's

Co-Disposal - Pkg. F = 1690 (1060 Concept B, C-D Pkgs.)
- Pkg. G = 1120 (280 Concept A, C-D Pkgs.)
(1340 Total C-D Pkgs.)

The Type F & G pkgs. will be individually dried and stored in a vault, conc. module, or cask storage system, then individually transported to the repository; i.e. placed in large MPC size overpack, once received at the repository.

The following larger fuels will be placed in Type G pkgs:

NBSR - 880
EBR-II - 121
TRR - 29
Slowpoke - 8
KMRR - 84
1121

Direct Disposal - Alts.

- 1) Use more small dia MPCs -
 Most applicable to MTR, which represents
 ~70% of the total and asys. with low kg/asy.,
 such that the repository kg U235 limit/ pkg.
 does not apply.

Small dia MPC capacity ~ 192- 240 MTR asys.
 (12 PWR cells)

MTR groups 5 & 6 (LEU, 287 and 172 asys.)
 allow caps in the 192-240 asy. range,
 but only 9 B pkgs. are required

Thus, use of small dia. MPC is not prudent
 for direct disposal, due to repository
 kg/pkg limits.

- 2) Use a mix of A/B pkgs. & co-disposal concepts A & B

This is a prudent concept to consider:

Egs:

Bs - Must have for KMAR/NRU, EBRII, TRR, & Slowpoke fuel

C-D - Concept A -Type G pkgs. - could be used in lieu of B pkgs.
 for the longer fuel

C-D - Concept B - Type F pkgs. work good for HFIR fuel

- 3) Use the max. no. of Type F pkgs.
 These short packages could possibly be loaded at the sending site, thus,
 avoid handling at SRS if a dual purpose cask (transportation and storage)
 is used. An alternate approach would include using the cask for transportation
 of the Type F pkg. then transferring the pkgs. to another storage mode at SRS.

Attachment IID-4

Criticality Analysis for Interim Storage and Transportation

Summary of NCS Analysis for Terry Bradley
D. Kent Parsons -- 6 March 1996
Document ESH-6-96-086
Los Alamos National Laboratory

In connection with the Spent Nuclear Fuel Storage and Disposal Assessment team, I have analyzed quite a few scenarios for the interim storage and ultimate disposal of DOE Al-clad spent nuclear fuel.

A very conservative assumption underlies all of this work. The fuel has been assumed to be fresh, i.e., no credit has been taken for burnup. Of course, if credit could be taken for burnup, then the limitations discussed below could be significantly relaxed. Burnup not only depletes the U-235 content of the elements, but it also builds up neutron poisons in the fuel.

For the purposes of fuel handling under water and for the theoretically possible (but highly unlikely) water flooding accident subsequent to drying and packaging, this analysis has primarily considered intact fuel bundles immersed in water. Without full water immersion, there are no practical criticality concerns for intact spent nuclear fuel elements of any variety in any size cask.

As a result of these two fundamental assumptions, the spent nuclear fuel is being characterized as having the same potential for criticality as a freshly loaded beginning of life research reactor core.

My basic fuel element is the classic 3" by 3" by 2 feet long MTR fuel element. I have conservatively assumed 450g of U-235 in each element. I then added enough U-238 to make the U-235 enrichment 94%. Of all the DOE spent nuclear fuel from both domestic and foreign sources, this model conservatively models the great majority of the material. Exceptions to this rule include any involute fuel (like HFIR or RHF) and the small number of "dense" MTR fuel elements in France. These dense MTR elements have from between 600 and 800g of U-235 in them. These exceptional fuels must be evaluated separately.

In my model, I assume that the MTR fuel element can be homogenized by simple volume weighting. The justification for this practice is that the fuel plates are thin and thus the U-235 does not self-shield in thermal neutron energies very much. In fact, homogenized HEU is always more reactive than heterogeneous HEU. So this homogenization adds yet another level of conservatism to the analysis. In addition, the conservatism of homogenization is the greatest when the fuel is the most heavily loaded.

My primary calculational tool was the DANT series of discrete ordinates codes. I used the well-known Hansen-Roach multigroup neutron cross sections. Because the DANT codes are deterministic, search calculations can be easily made to determine critical masses. I have also used the continuous energy version of the Monte Carlo code MCNP to verify some of my results. The independence of the calculational methods and the

cross section data between DANT and MCNP establishes the credibility of the results.

I was asked by Terry Bradley to consider arranging arrays of MTR fuel elements without neutron poisons as a starting case. The objective was to see how many of these assemblies could be packed together in some sort of a storage container. Under the assumptions of "fresh" fuel and water immersion, the answers, as determined by TWODANT were:

	HEU fuel at 450g U-235	LEU fuel at 450g U-235
Crit.Mass	2.99kg of U-235	3.67kg of U-235
No. of Elements.	6.6	8.2

This analysis assumed a 1-high square array with the individual fuel elements actually touching each other. The first result was verified with MCNP and a k_{eff} of 0.996 ± 0.003 was obtained for the critical case.

If the MTR elements were only loaded to 300g of U-235 per assembly, then the results would have been:

	HEU fuel at 300g U-235	LEU fuel at 300g U-235
Crit.Mass	2.40kg of U-235	2.91kg of U-235
No. of Elements.	8.0	9.7

These calculated critical masses compare well, in a conservative sense, with the following experimental measurements (see references 1 and 2) of fresh MTR-like fuel elements:

	HEU fuel at 168g U-235	HEU fuel at 300g U-235
Crit.Mass	2.5kg of U-235	3.5kg of U-235
No. of Elements	15	11

Thus, we can conclude that for the assumptions of no burnup credit and full water immersion, relatively few of these MTR elements can be arranged together without poisons.

For the poisoned cases, I have analyzed two scenarios involving storage casks in PWR configurations (i.e., 4 MTR elements in a 9" by 9" PWR assembly basket) and one scenario with a storage cask in a BWR configuration (i.e., one spent fuel element in a 6" by 6" BWR assembly basket). Due to the lack of time, I performed a scoping k_{inf} analysis. k_{inf} is different than k_{eff} . k_{inf} is the multiplication factor for an infinite system with no neutron leakage at all. k_{eff} is the multiplication factor for a finite system which has neutron leakage.

The rationale is that if k_{inf} is < 1.0 for a single unit, then any array of identical units is also subcritical. In practice, if k_{inf} for a single unit is < 1.0 , then any finite array of identical units is substantially sub-critical. Thus a complex configuration can be reduced in a conservative manner to

the analysis of a single simple unit cell.

For poisons, I used a 1/4" thick sheet of Boral or a 1% natural boron by weight stainless steel in 1/4" plates between the elements. I also accounted for the water between the fuel elements in the space inside the basket.

For the 450g MTR elements arrayed together with 1/4" borated stainless steel plates in between, k_{inf} is 1.055. So no upper limit is possible without further analysis. However, due to the preponderance of less heavily loaded fuel for which $k_{inf} < 1.0$ (as discussed below), it should be a simple matter to mix elements such that for any size cask all available space is utilized.

If 300g elements are stored with borated stainless steel, then k_{inf} is 0.891. Thus, any but the most heavily loaded MTR elements may be stored in borated steel baskets in PWR casks without mass or number of element limitations.

If 1/4" sheets of Boral are used instead of the borated stainless steel, then the k_{inf} is 0.983 for the 450g elements. MCNP gives 0.981 ± 0.003 . Hence, there is no mass or number of element limitation for the intact 450g fuel elements when stored in 2 by 2 arrays in commercial PWR baskets lined with Boral.

For the 6" by 6" Boral lined BWR slots, up to 1000g of U-235 in Al-clad fuel may be stored in every 2 feet of height. The k_{inf} of this system is 0.991. These BWR slots should be useful in storing any spent fuel elements or assemblies which are larger than 3" by 3".

Thus, we can conclude that even with the assumption of no burnup credit and full water immersion, there are no serious limitations on the storage of intact MTR and MTR-like fuel elements in poisoned storage casks. While some of the k_{inf} values quoted above may appear to be higher (i.e., closer to 1.00) than usual, we are confident that more expensive (and time-consuming) calculations for finite storage casks will result in values of $k_{eff} <$ any limit (e.g., 0.95 minus some uncertainty such as 0.02, as currently required in 10 CFR 60) proscribed by regulatory bodies.

All of these calculations illustrate a important characteristic of spent nuclear fuel storage. The repository, or any other storage facility, is very likely to be more limited by volume or space shortages than it will be by criticality restrictions. This is even more true if some of the unnecessary conservatism in the criticality analysis can be relaxed.

References:

(1) J.K.Fox and L.W.Gilley, "Critical Experiments with Arrays of ORR and BSR Fuel Elements," Neutron Physics Annual Progress Report for the period ending September 1, 1958, pp. 34-36, ORNL-2609, October 1958.

(2) E.B.Johnson and R.K.Reedy, Jr., "Critical Experiments with SPERT-D Fuel Elements," ORNL-TM-1207, July 1965.

Attachment IID-5

Summary of Safeguards and Security Requirements

B. Safeguards and Security Requirements During Interim Storage and Transport of Highly Enriched Aluminum Clad Spent Nuclear Fuel Cask.

It is Department's policy to comply with the safeguards and security requirements of Nuclear Regulatory Commission (NRC), Department of Transportation (DOT) and DOE during Interim Storage and Transport of Highly Enriched Aluminum Clad Spent Nuclear Fuel Cask.

SNF at SR is classified as Category III material in accordance with the requirements of the DOE Order 5633.3B, "Control and Accountability of Nuclear Materials" Dated 09/07/94. The SNF will be bundled such that load will be self protecting. The DOE Order 5633.3B prescribes the Department's minimum requirements and procedures for control and accountability of nuclear materials at DOE-Owned and leased facilities and DOE-owned nuclear materials at other facilities which are exempt from licensing by the Nuclear Regulatory Commission.

The dry storage of SNF at SR shall comply with the physical protection requirements of NRC as established in Title 10 Code of Federal Regulations 72 Subpart H, "Physical Protection establishes physical protection requirements for the Independent Storage of Spent Nuclear Fuel and high-Level Radioactive waste.

The physical protection of SNF casks during transportation shall comply with the NRC requirements in Title 10 Code of Federal Regulations 73.37, " Requirements for Physical Protection of Irradiated Reactor Fuel in Transit," which identifies procedures for protection of licensee shipments of irradiated reactor fuel.

In addition, the storage and transportation of SNF shall also comply with the DOE Order 5632.1C, "Protection and Control of Safeguards and Interests", dated 07/15/94 establishes policy for protection and control of special nuclear material, vital equipment, classified matter, Departmental property and facilities, and unclassified irradiated reactor fuel in transit and DOE Order 460.2, "Material Transportation and Packaging management" establishes policies and requirements to supplement applicable laws, rules, regulations, and other DOE Orders for materials transportation and packaging operations.

Attachment IID-6

Drying & Pressure Buildup Issues for Aluminum-Clad Spent Nuclear Fuel

Drying & Pressure Buildup Issues for Aluminum-Clad Spent Nuclear Fuel

R. G. Ballinger

6/1/96

1.0 Introduction

The spent nuclear fuel will undergo several processing and/or storage steps between receipt of the fuel and ultimate disposition in a repository. During operation and storage the fuel will undergo general and, possibly, localized corrosion such as pitting. Indeed, fuel currently being stored at Savannah River has suffered corrosion, general and pitting, to some degree. During storage of the spent nuclear fuel (SNF) in closed containers a slow pressure buildup will occur in the storage container due to the radiolytic decomposition of any water present (free or bound) and the reaction of free water with the cladding material. The ultimate gaseous reaction products will be hydrogen and oxygen. The oxygen will react with any metal surfaces present with the net result that the storage container will develop a hydrogen rich environment. The maximum pressure attained during closed storage will be determined by the amount of water present at the time of closure in combination with any fill gas that is used. The RATE of pressure buildup will be determined by the reaction rate of any free water with the fuel which will be controlled by the rate of release of bound water due to radiolysis. The limiting step in the process will depend on the relationship between the rate of release of bound water due to radiolysis and the reaction rate with the cladding and other structural materials. In any case, the process will ultimately result in pressure buildup. Additional, immediate pressure buildup will occur if free water is present in the storage container due to the equilibrium vapor pressure at the storage temperature. Since the reaction of water to produce hydrogen is a one-for-one "trade" the pressure due to this process will add a constant value to the overall system pressure.

The minimization of pressure buildup and the limitation of the terminal pressure will be important design considerations for any of the storage and/or processing options for metallic fuel forms. Pressure buildup will be kept within design limits either by periodic venting of a closed storage vessel or by limiting the absolute terminal pressure to a value less than the design pressure. In either case an adequately accurate estimate of the total amount of water present as well as an estimate of the rate of reaction will be important.

The removal of water in the storage container will require some form of drying process. The water present in the storage container will be present in one of three general forms: (1) free water, (2) water of hydration for oxides present, and (3) chemisorbed water on all surfaces. The drying process can be expected to remove essentially all of the free water. Water which is in the form of hydrated oxides will decompose if the drying conditions are such that the hydrated oxide is unstable and sufficient time is allowed for the kinetics of decomposition to operate. The degree of removal of this form of water will depend on the stability of a particular oxide at the drying conditions. The removal of chemisorbed water will require drying at temperatures that far exceed the maximum allowable fuel storage temperature of 200C. Thus the pressure buildup during storage will be due to the decomposition of any hydrated oxides present after drying and the release of chemisorbed water.

As part of the evaluation of processing options the potential for pressure buildup was evaluated for the dry storage and direct disposal options. In fact, since all of the options will require some form of interim storage some form of drying will have to take place unless a vented interim storage option is elected. In this Appendix we present the results of an expanded study to evaluate the potential for pressure buildup during interim or permanent storage. As part of the evaluation an analysis was conducted to determine both the likely forms of corrosion product oxides that may be present but to evaluate the likelihood of the drying process resulting in the removal of some of the bound water. The results of this analysis are presented in Section 2.0. In Section 3.0 we present the results of the analysis of potential pressure buildup during storage. For this analysis the evolution of container pressure was modeled as a function of time. All sources of pressure buildup were included in the model. These included: (1) pressure buildup from the corrosion process, (2) radiolytic decomposition pressure buildup, (3) the effect of the presence of free water, and (4) the effect of temperature.

2.0 Oxide Stability

In this section we present the results of an analysis of the anticipated drying behavior for SNF.

2.1 The Drying Process

The drying process will result in the removal of any free water and bound water from hydrated oxides that may be unstable at the drying temperature and pressure. Figure 2.1-1 shows a schematic of the drying behavior for vacuum drying for the case where free water is present as well as corrosion product oxides.

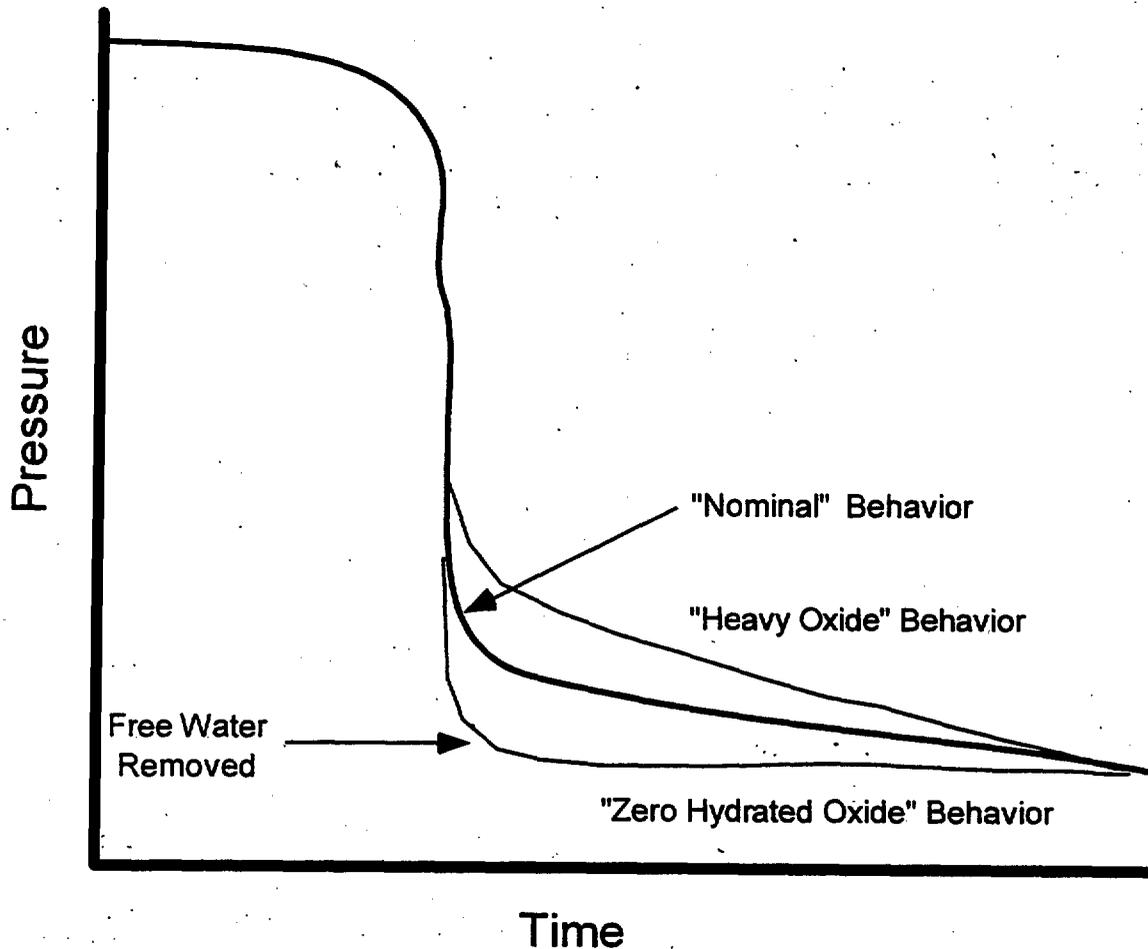


Figure 2.1-1 Schematic drying curve for case of only free water present.

There has been a significant amount of experience with commercial light water reactor fuel drying in preparation for dry storage. The schematic behavior above is characteristic of a situation where some retained oxide is present on the fuel. This retained oxide will act to entrain free water. In Figure 2.1-1 the curve labeled "Zero Oxide" Behavior is the behavior which is expected, and is very often seen, when the fuel surfaces are essentially free of oxide that contains any free water. Thus one observes that the pressure quickly drops to a very low pressure, corresponding to the limiting pressure of the pumping system, once all of the free water has been evaporated. On the other hand, if there is some amount of trapped, but unbound, water present one would expect behavior such as those represented by the curved which approach the "zero" free water point more gradually. The point to be made here is that there will be a point where all of the free water will be removed.

For the case where there is some amount of bound water present in hydrated oxides the pressure-time behavior will depend on the amount of, type, and stability of the oxides. Figure 2.1-2 shows a schematic of this type of behavior. Note in Figure 2.1-2 that the time axis is not drawn to scale. The time that it

takes for an equilibrium pressure to be established will depend on the amount and type of hydrated oxide present.

Figures 2.1-3 through 2.1-34 show the phase stability diagrams for oxides and hydroxides of uranium, aluminum and iron as a function of temperature. Figure 2.1-35 shows the equilibrium partial pressure of water over hydrated oxides and hydroxides of uranium, aluminum and iron as a function of temperature. Figure 2.1-36 shows the free energy for the decomposition reactions for hydrated oxides and hydroxides of uranium, aluminum and iron as a function of temperature at 1 atmosphere. The data presented in these figures is for the pure hydrated oxides. It is likely that more complex compounds with more variable chemistry will be present. However, it is unlikely that the behavior of the more complex chemistries will be significantly different unless a significantly different hydrated oxide form is present.

Based on the analysis conducted it is anticipated that the primary hydrated oxide present on the fuel after drying at 50C and a pressure of .001 atm, the anticipated drying conditions, will be the aluminum oxide mono-hydrate.

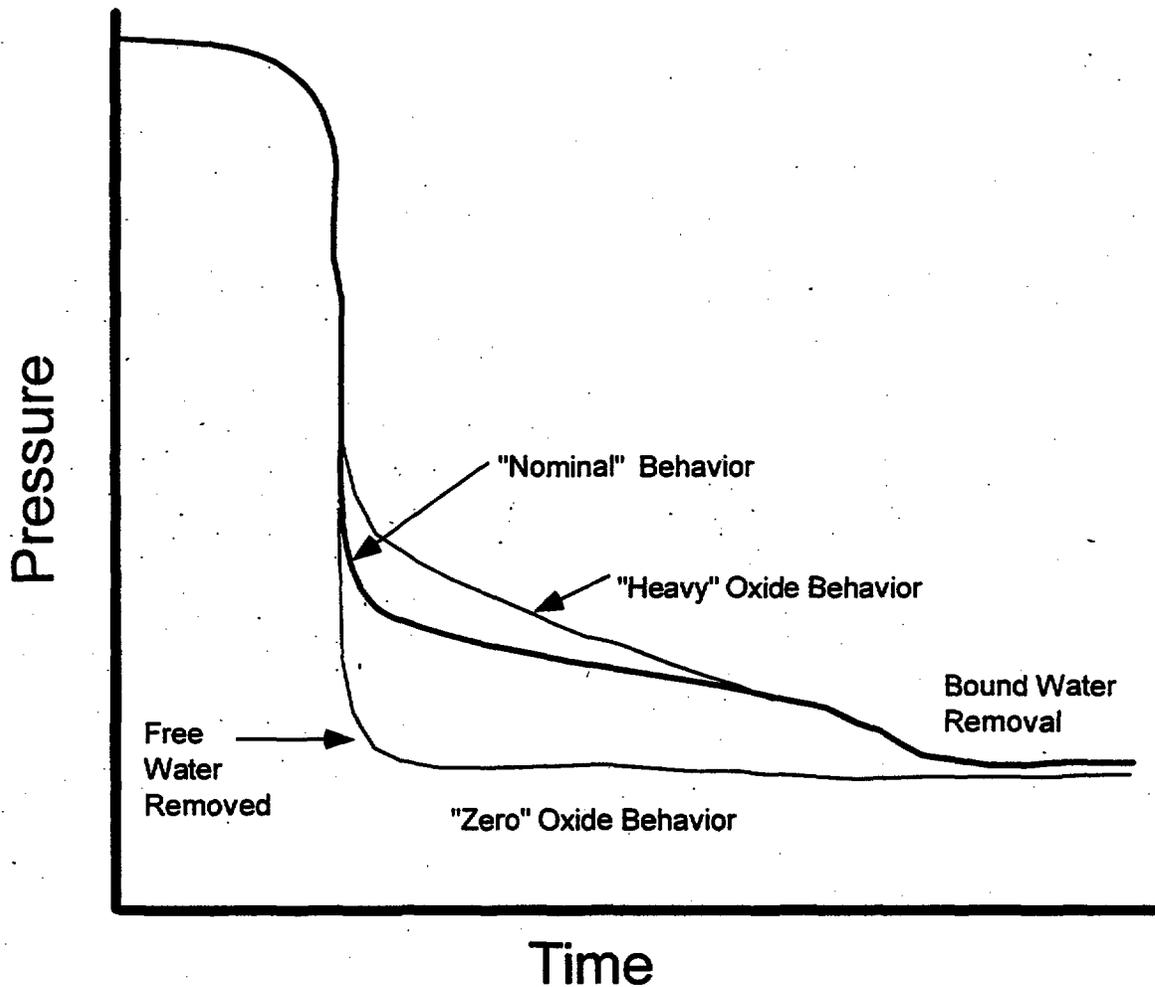


Figure 2.1-1 Schematic drying behavior for the case where hydrated oxides are present. Note that the time axis is not drawn to scale.

3.0 Pressure Buildup

During the drying process all free water will be removed from the fuel. However, some bound water will still remain either in the form of hydrated oxides or in the form of chemisorbed water. Drying data for sludge/oxides from the K-Basin in Hanford indicates that the chemisorbed water is not removed until exposures at very high temperatures, on the order of 900C. The remaining water will eventually be released as a result of radiolytic decomposition to form hydrogen and oxygen. The oxygen released will be consumed in the formation of other oxides as a result of reaction with fuel, canister and other metal surfaces. However, the hydrogen will remain and will combine with the fill gas to result in an increase in the pressure in the storage container. This process has been modeled and this section presents some of the results of this analysis.

Figures 3.0-1 through 3.0-3 show the evolution of conditions in the storage container for typical fuel element loading (32 elements), free volume (9.4 cubic feet), and oxide thickness (0.0002 in (0.0005 cm)). The ultimate pressure in the storage container is just under three atmospheres. Data related to fuel oxide thickness, and other parameters were obtained from a report by Peacock, et. al.[1]. The fuel gamma heat loading was taken as 450 watts. This heat loading is considered an upper bound for all fuel with the exception of HFIR fuel.

The rate of pressure buildup is gradual with the terminal pressure being reached after approximately 50,000 hours for this case. For the case of HFIR fuel the terminal pressure will be reached in approximately 1000 hours.

Figure 3.0-4 shows the terminal pressure in the storage container as a function of fuel surface oxide thickness for several storage container free volumes. The storage temperature for this figure is 200C. The oxide thickness ranges modeled bracket the expected fuel conditions [1].

4.0 Conclusions

Based on this analysis it has been concluded that fuel drying will be adequate to prevent excessive pressure buildup during dry storage.

References

1. Peacock, H. B., Sindelar, R. L., Lam, P. S., and Murphy, T. H., "Evaluation of Corrosion of Aluminum-Base Reactor Fuel Cladding Materials During Dry Storage (U), WSRC-TR-95-0345, November, 1995.

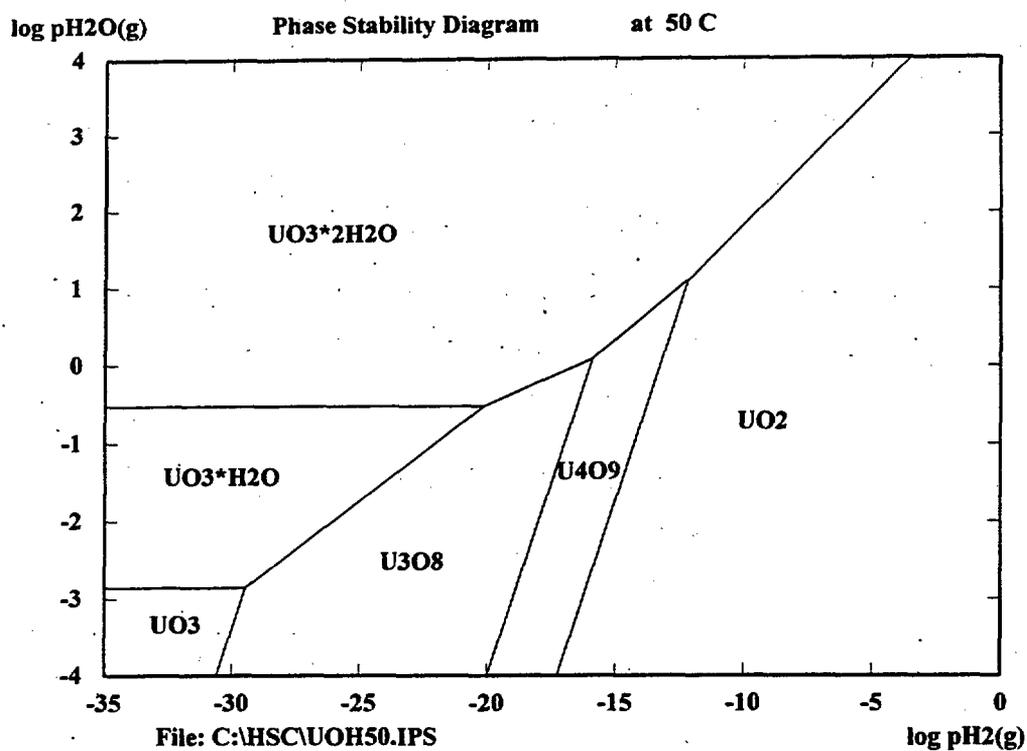


Figure 2.1-3 Phase stability for the uranium oxide-water-hydrogen system at 50C.

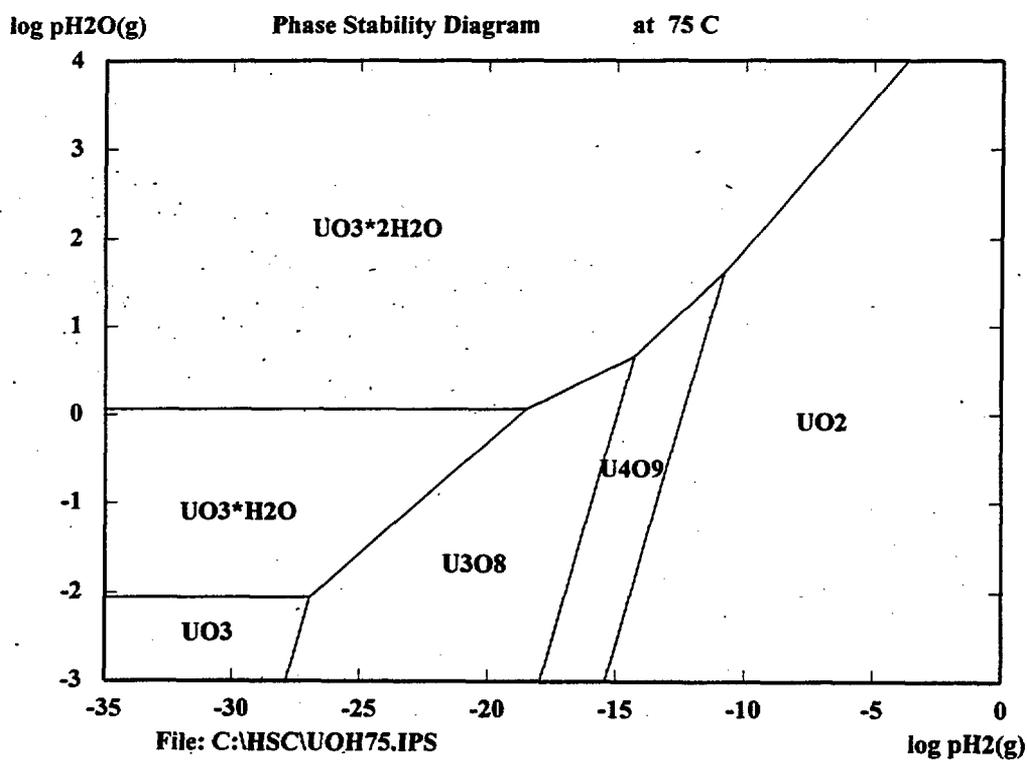


Figure 2.1-4 Phase stability for the uranium oxide-water-hydrogen system at 75C.

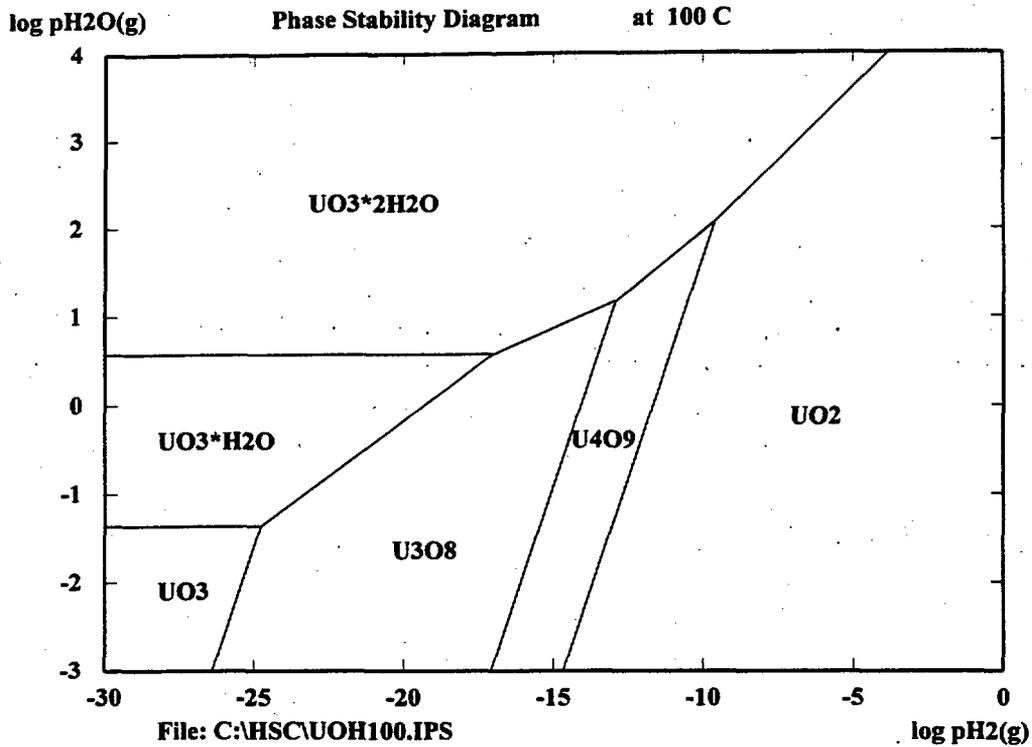


Figure 2.1-5 Phase stability for the uranium oxide-water-hydrogen system at 100C.

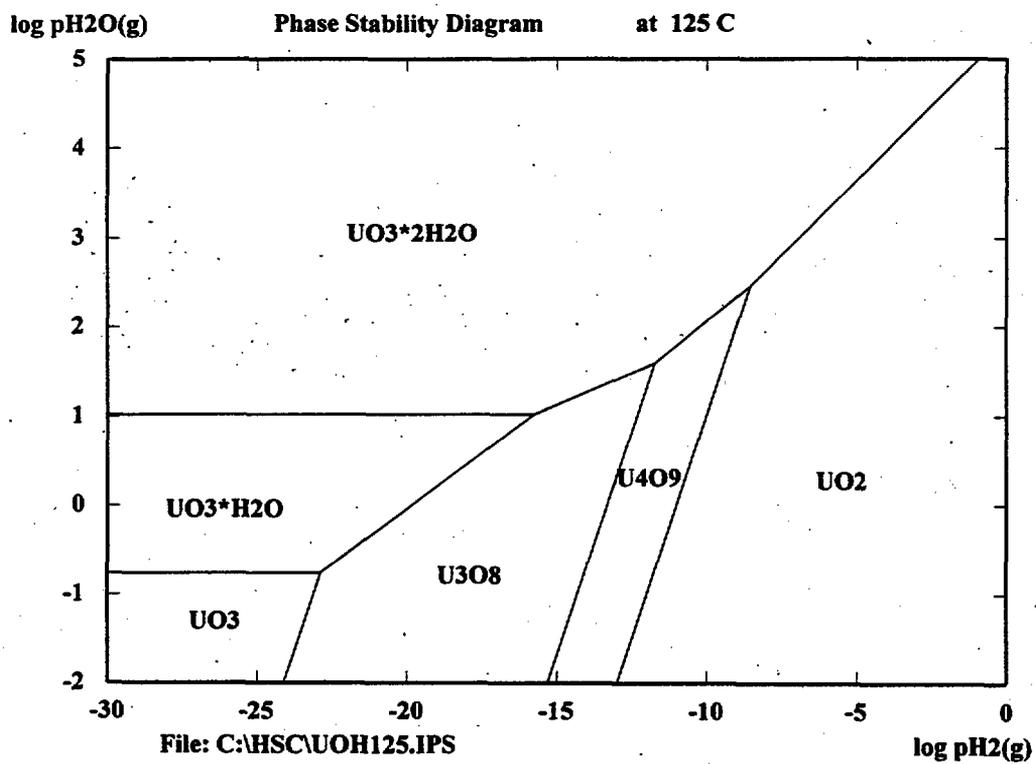


Figure 2.1-6 Phase stability for the uranium oxide-water-hydrogen system at 50C.

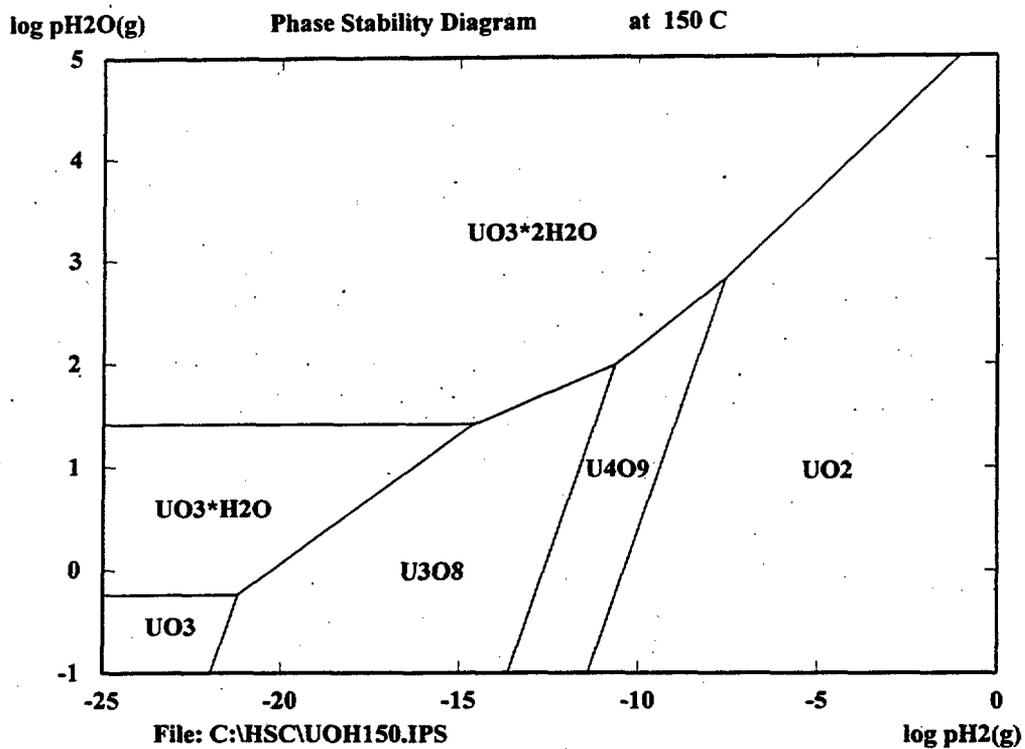


Figure 2.1-7 Phase stability for the uranium oxide-water-hydrogen system at 150C.

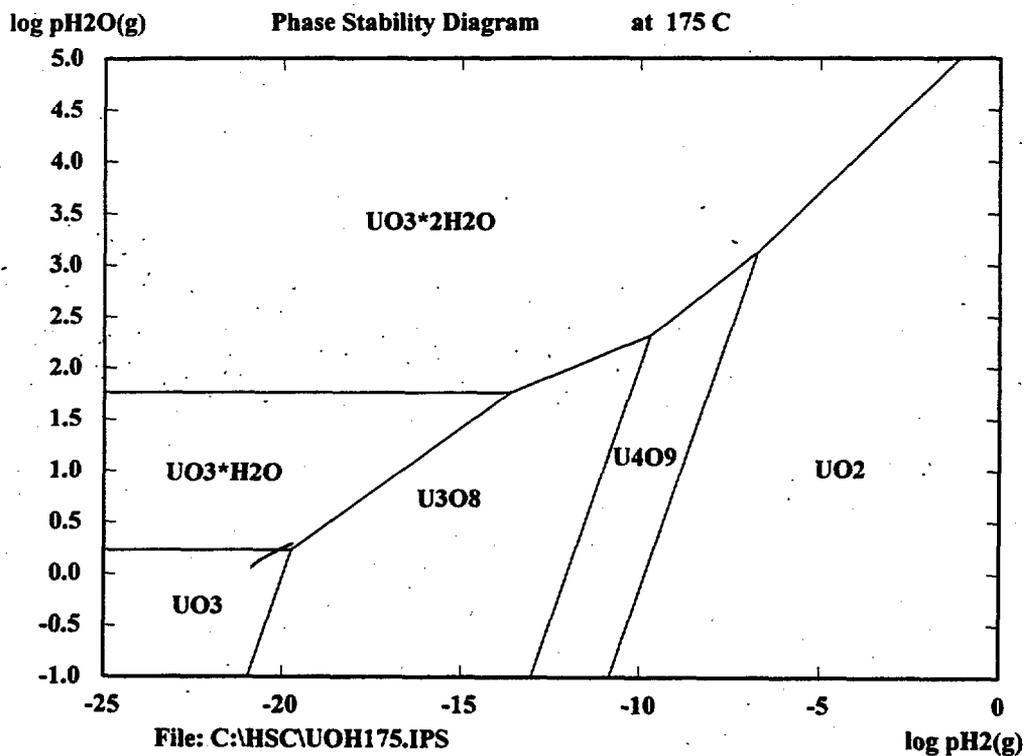


Figure 2.1-8 Phase stability for the uranium oxide-water-hydrogen system at 175C.

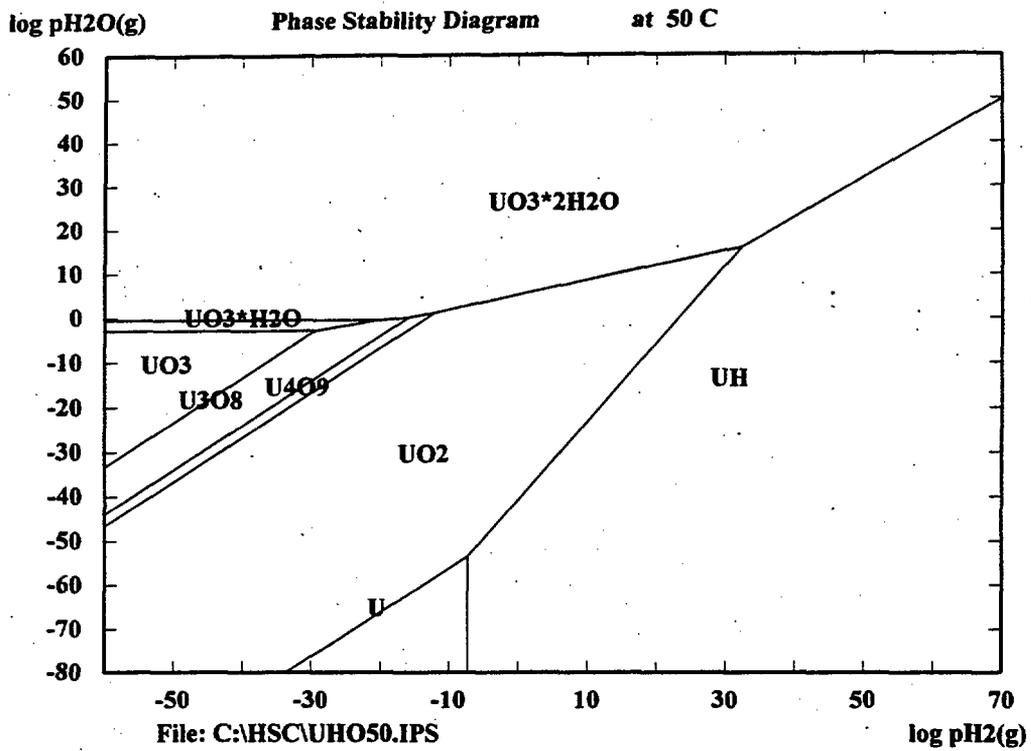


Figure 2.1-9 Phase stability for the uranium-water-hydrogen system at 50C.

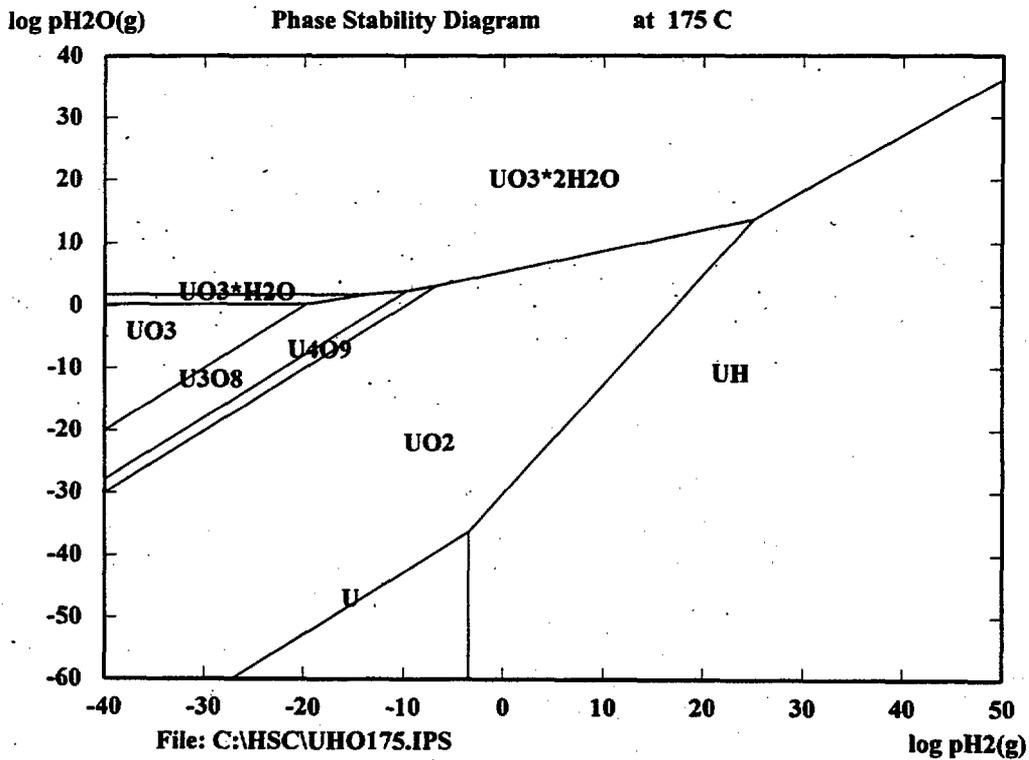


Figure 2.1-10 Phase stability for the uranium-water-hydrogen system at 175C

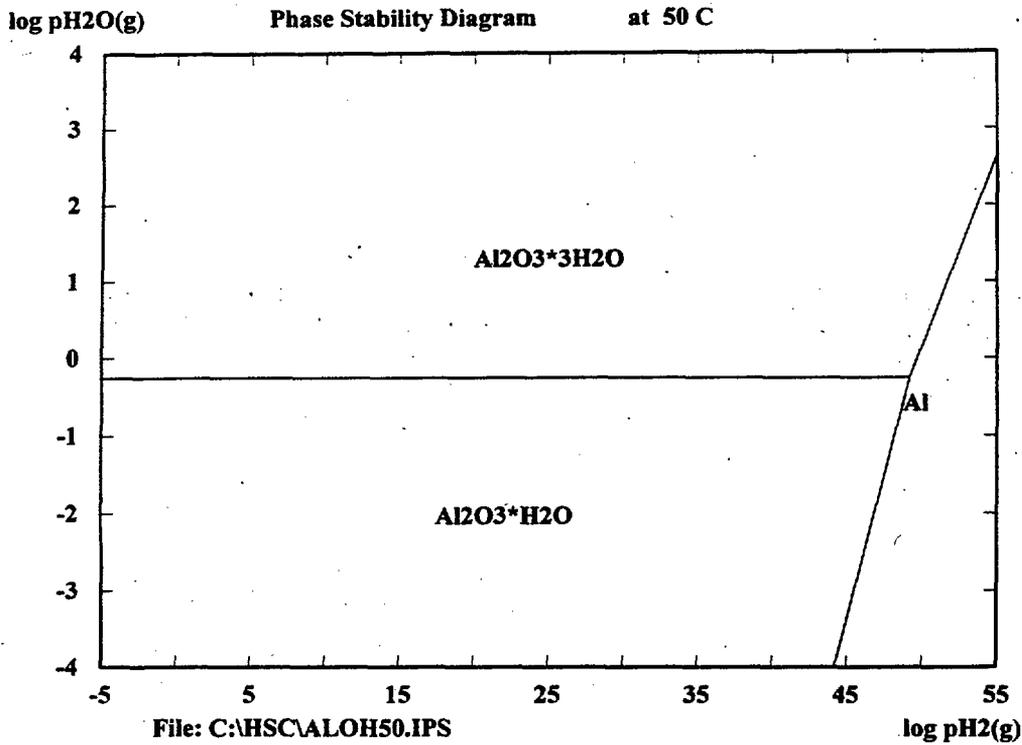


Figure 2.1-11 Phase stability for the aluminum oxide-water-hydrogen system at 50C.

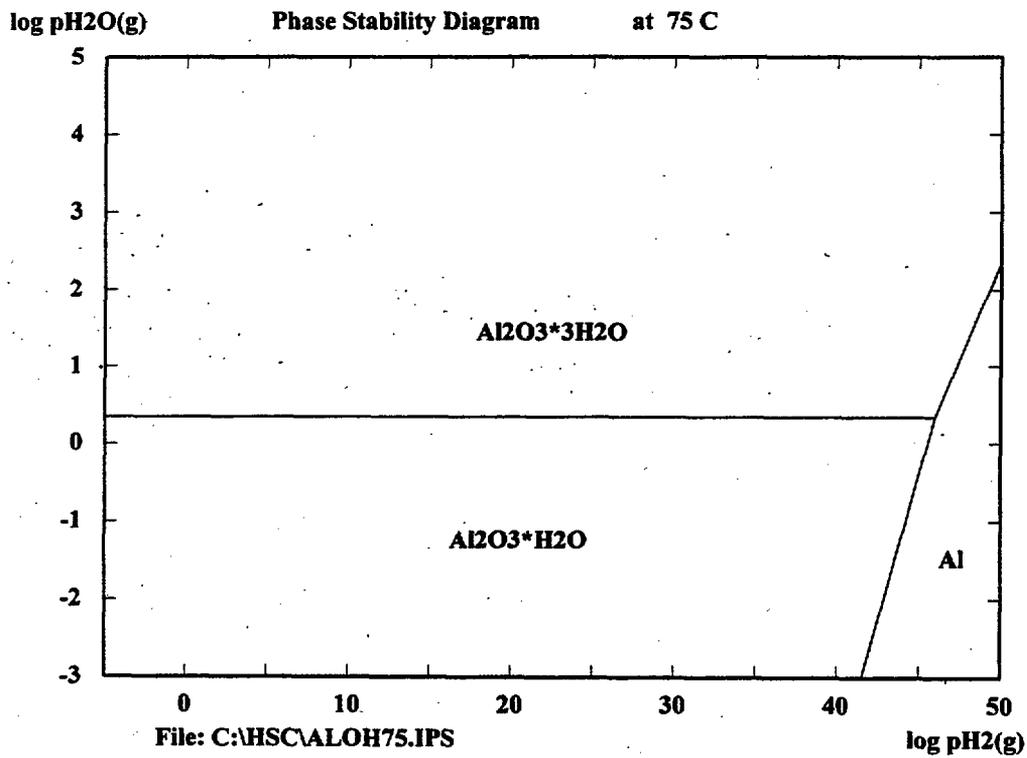


Figure 2.1-12 Phase stability for the aluminum oxide-water-hydrogen system at 75C.

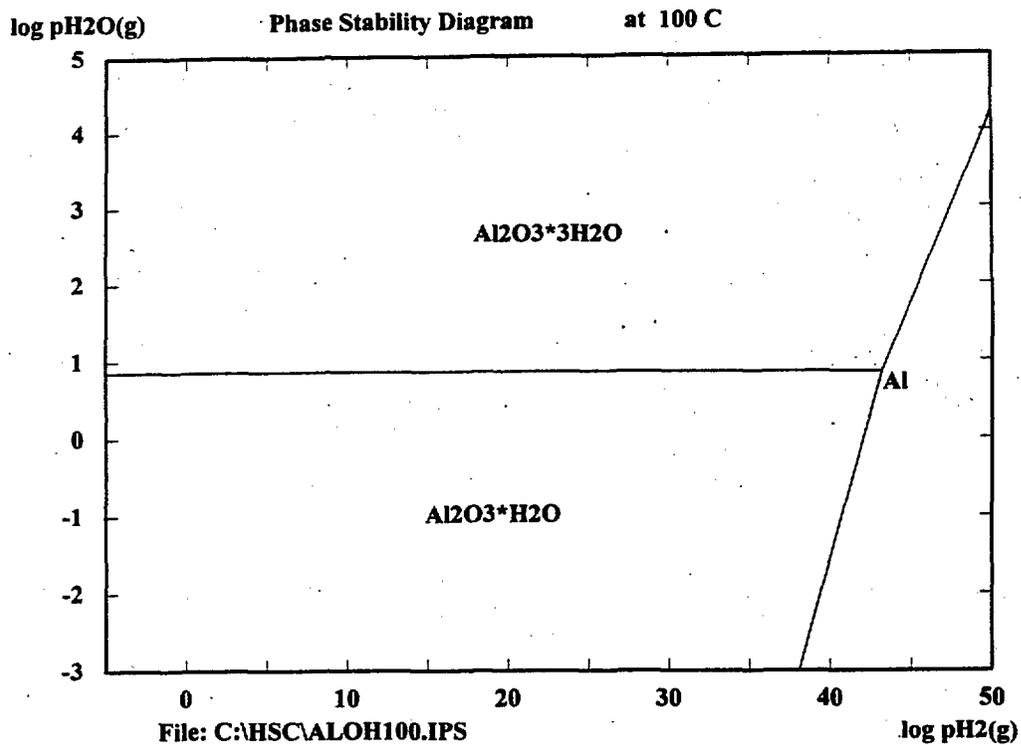


Figure 2.1-13 Phase stability for the aluminum oxide-water-hydrogen system at 100C.

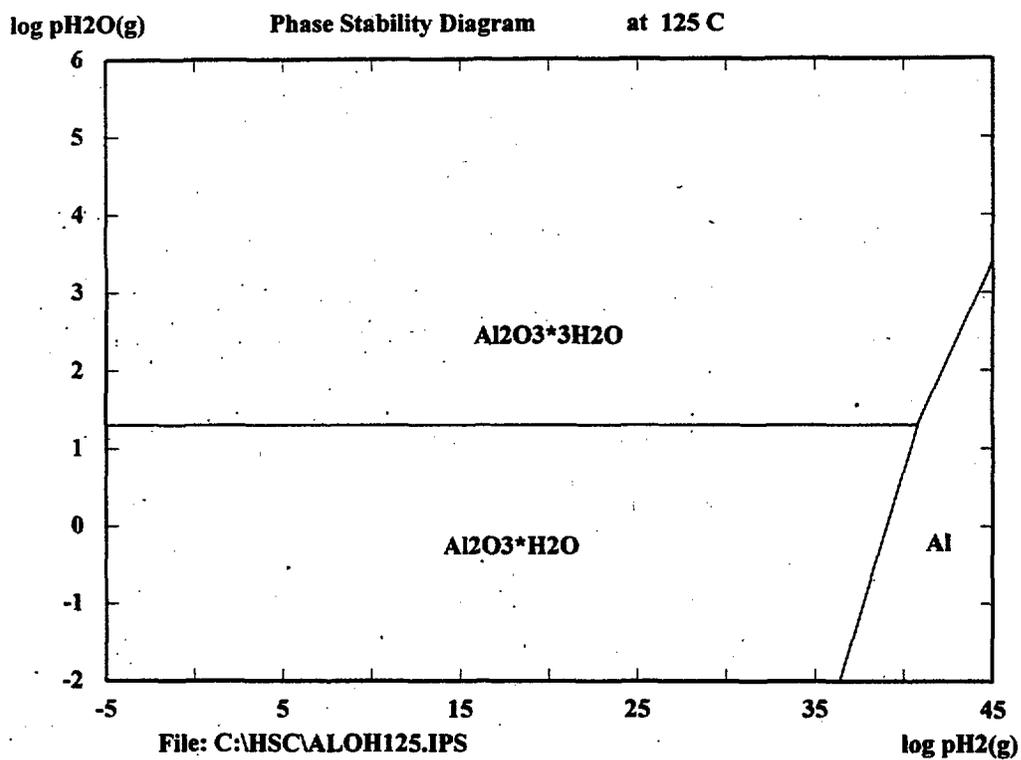


Figure 2.1-14 Phase stability for the aluminum oxide-water-hydrogen system at 125C.

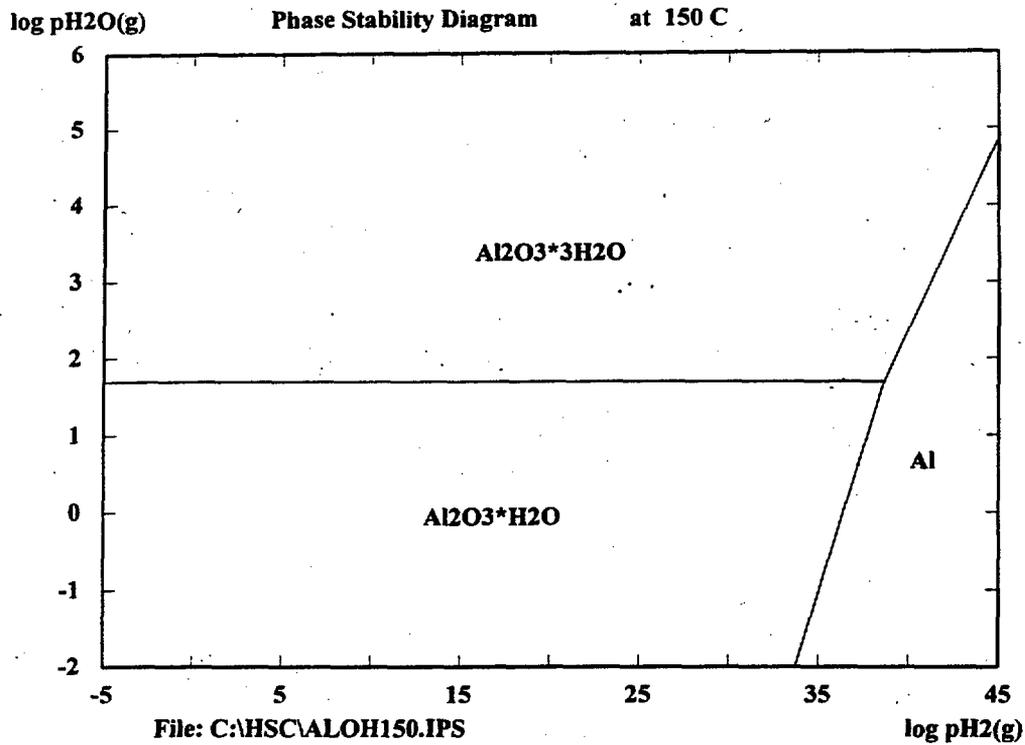


Figure 2.1-15 Phase stability for the aluminum oxide-water-hydrogen system at 150C.

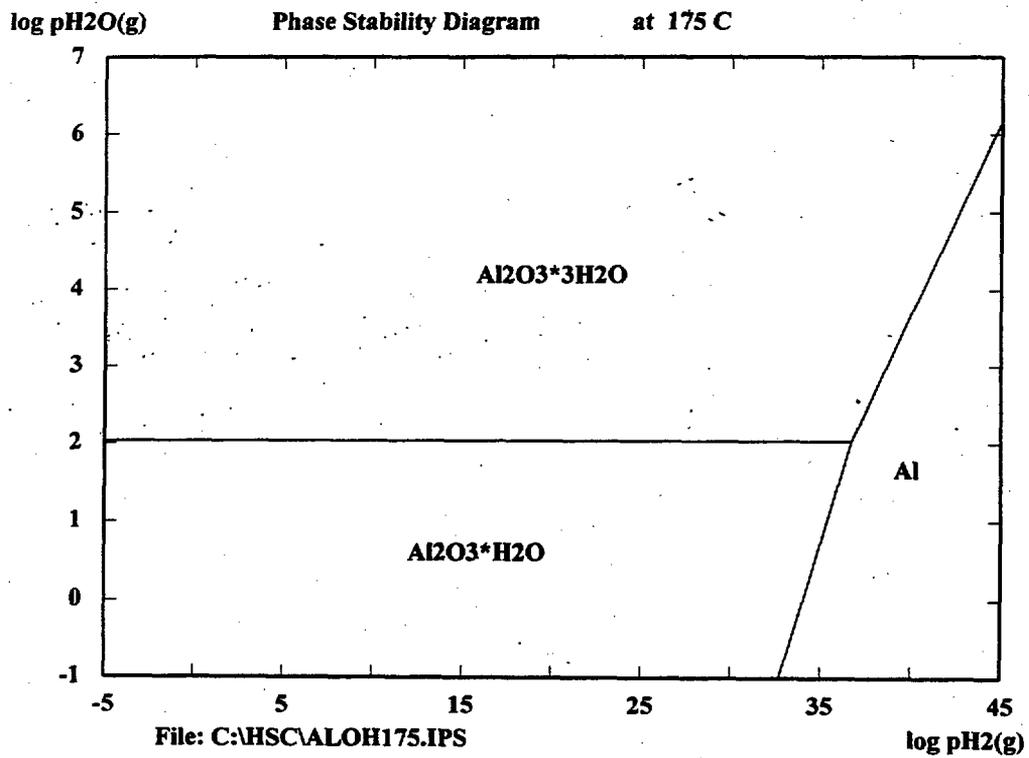


Figure 2.1-16 Phase stability for the aluminum oxide-water-hydrogen system at 175C.

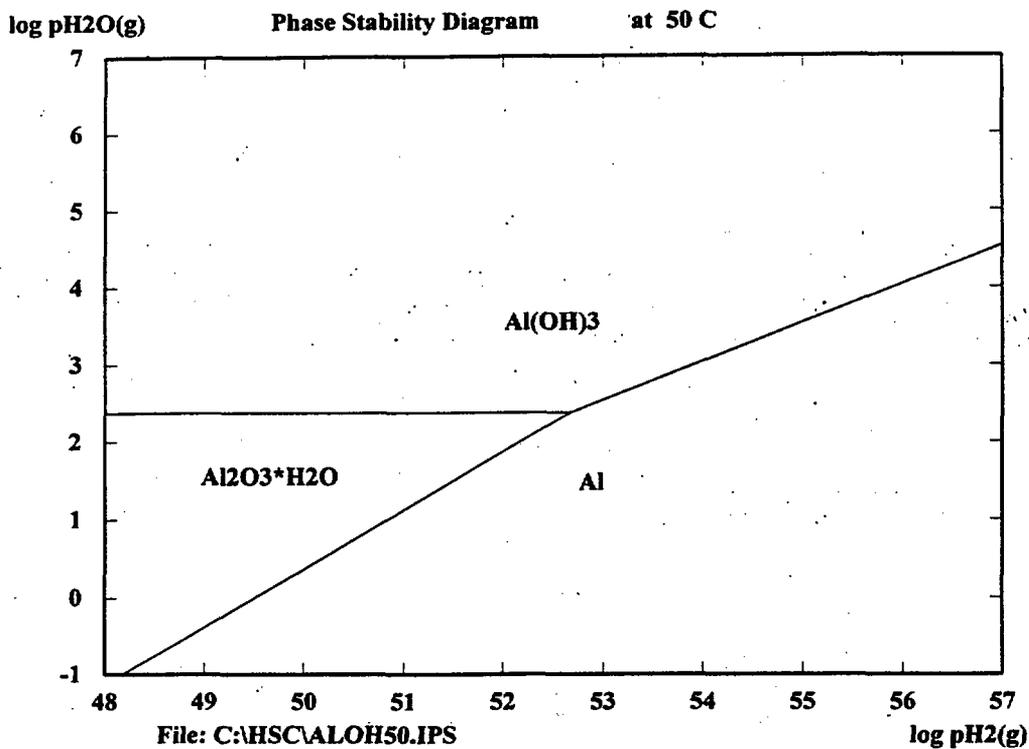


Figure 2.1-17 Phase stability for the aluminum hydroxide-water-hydrogen system at 50C.

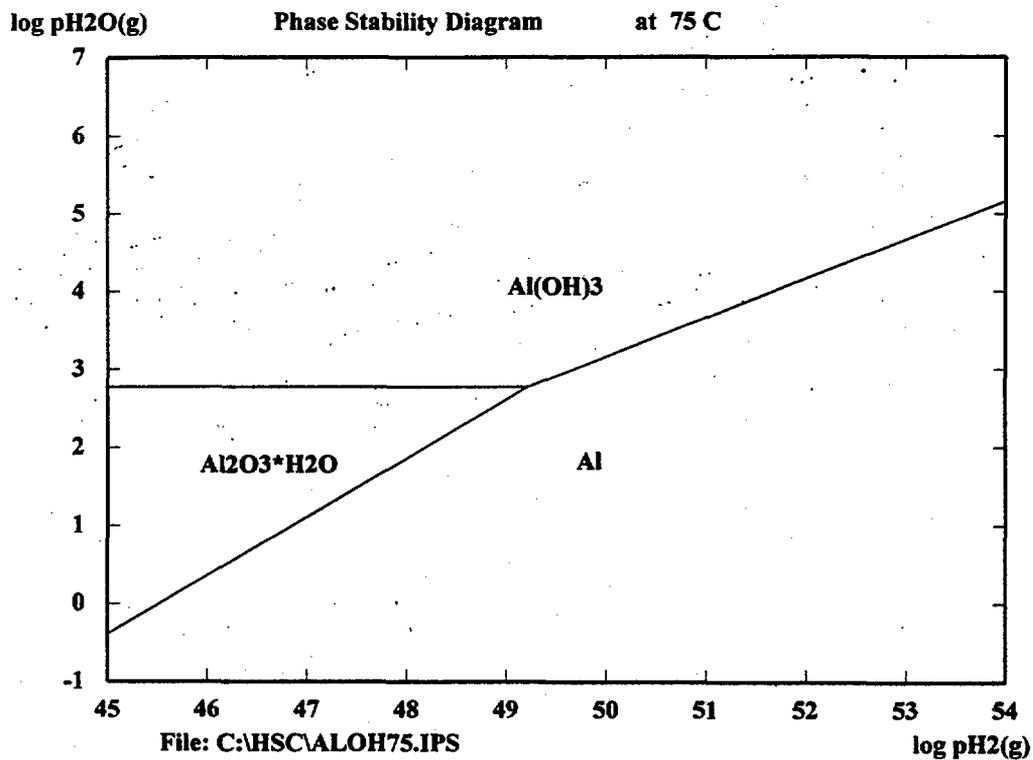


Figure 2.1-18 Phase stability for the aluminum hydroxide-water-hydrogen system at 75C.

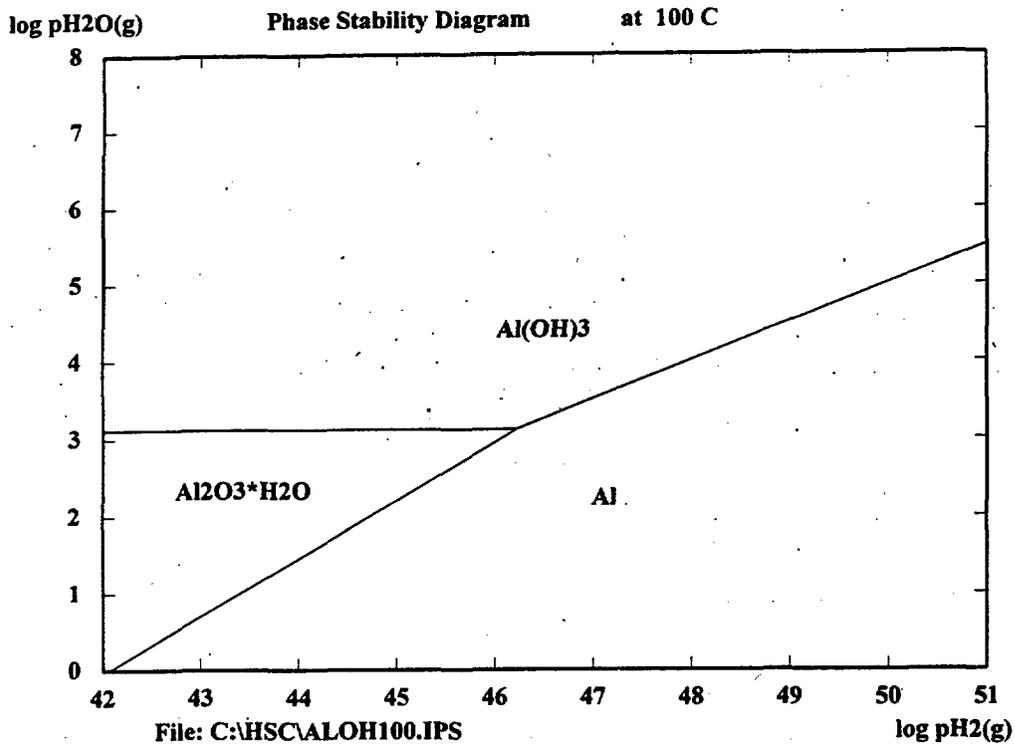


Figure 2.1-19 Phase stability for the aluminum hydroxide-water-hydrogen system at 100C.

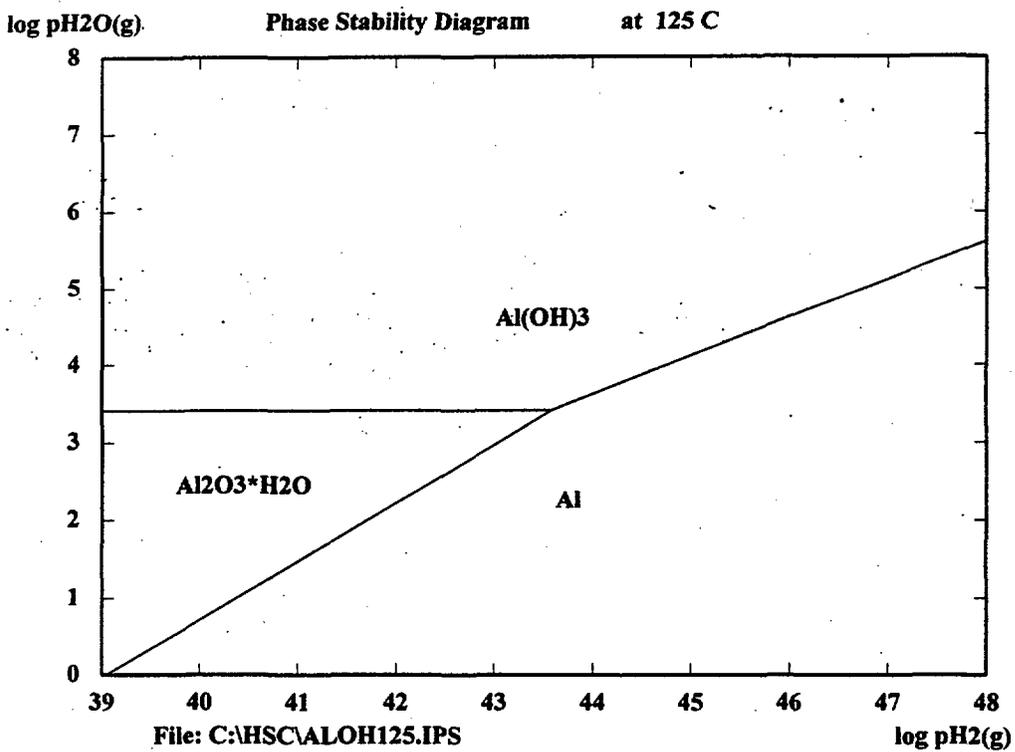


Figure 2.1-20 Phase stability for the aluminum hydroxide-water-hydrogen system at 125C.

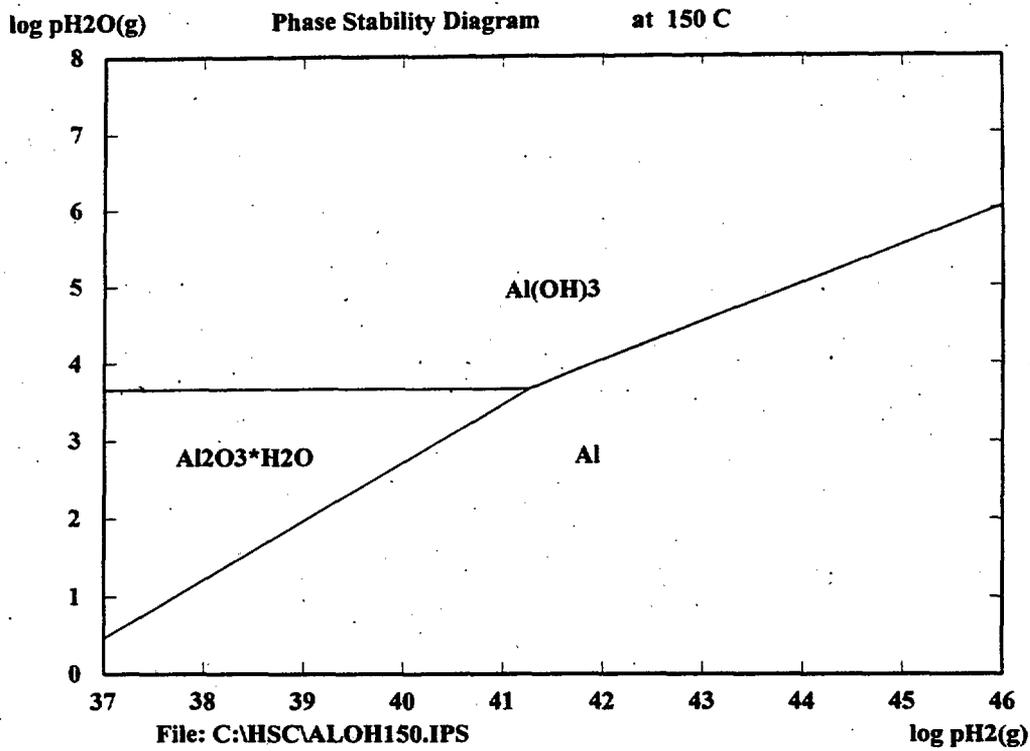


Figure 2.1-21 Phase stability for the aluminum hydroxide-water-hydrogen system at 150C.

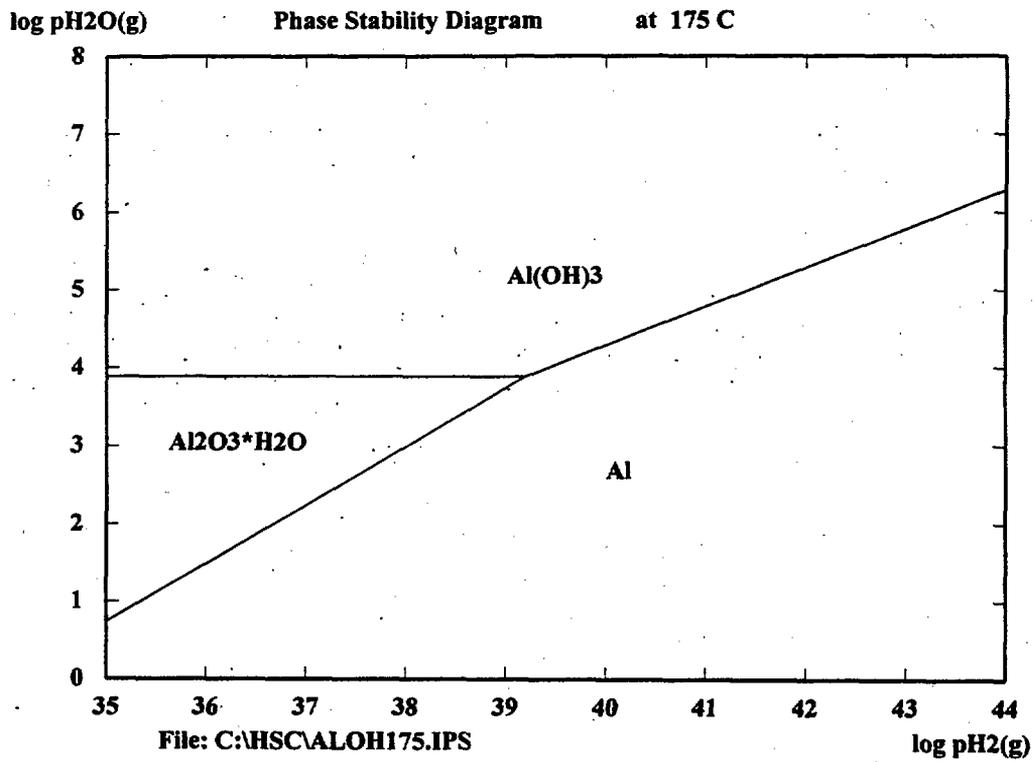


Figure 2.1-22 Phase stability for the aluminum hydroxide-water-hydrogen system at 175C.

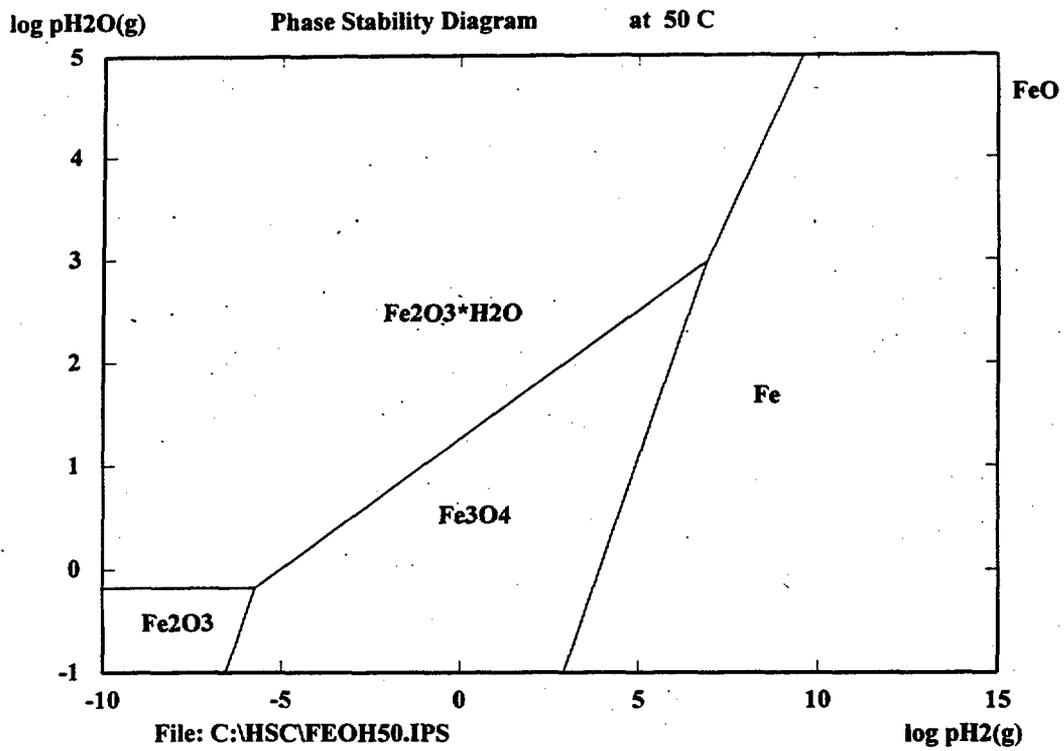


Figure 2.1-23 Phase stability for the iron oxide-water-hydrogen system at 50C.

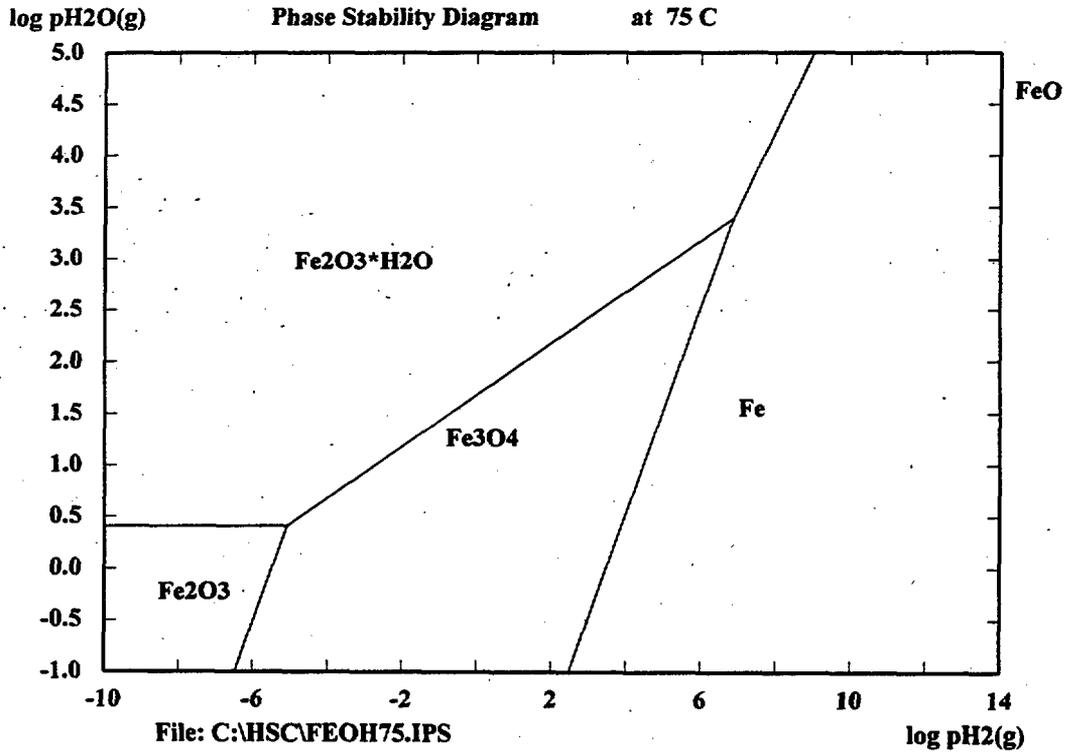


Figure 2.1-24 Phase stability for the iron oxide-water-hydrogen system at 75C.

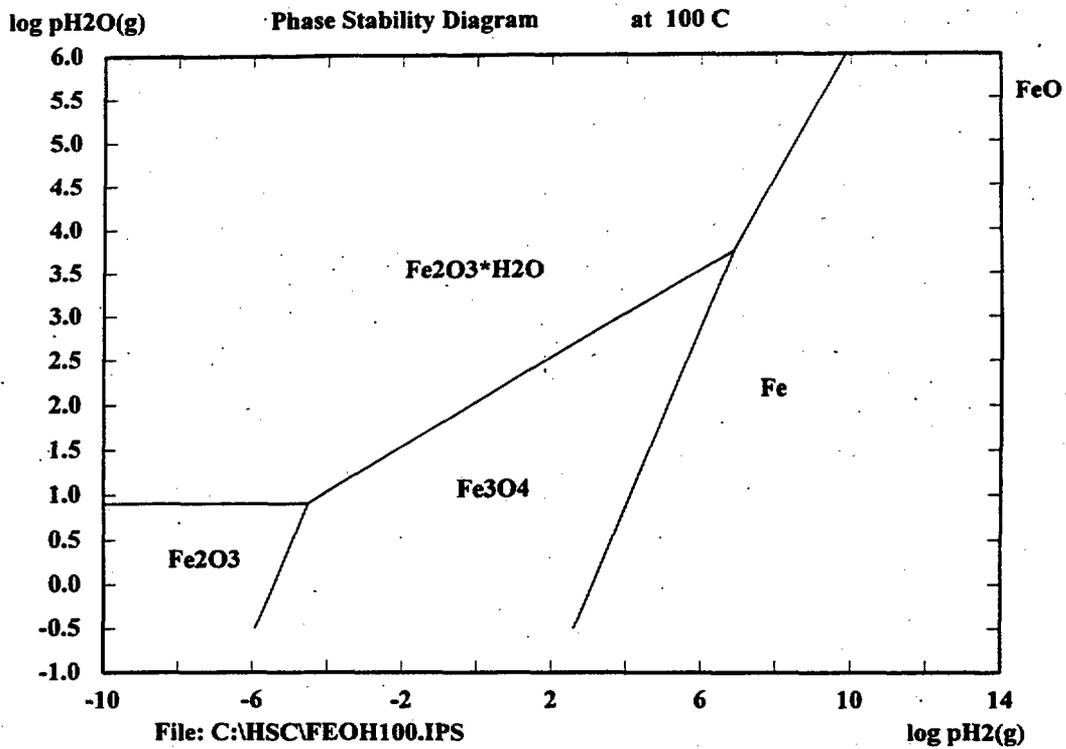


Figure 2.1-25 Phase stability for the iron oxide-water-hydrogen system at 100C.

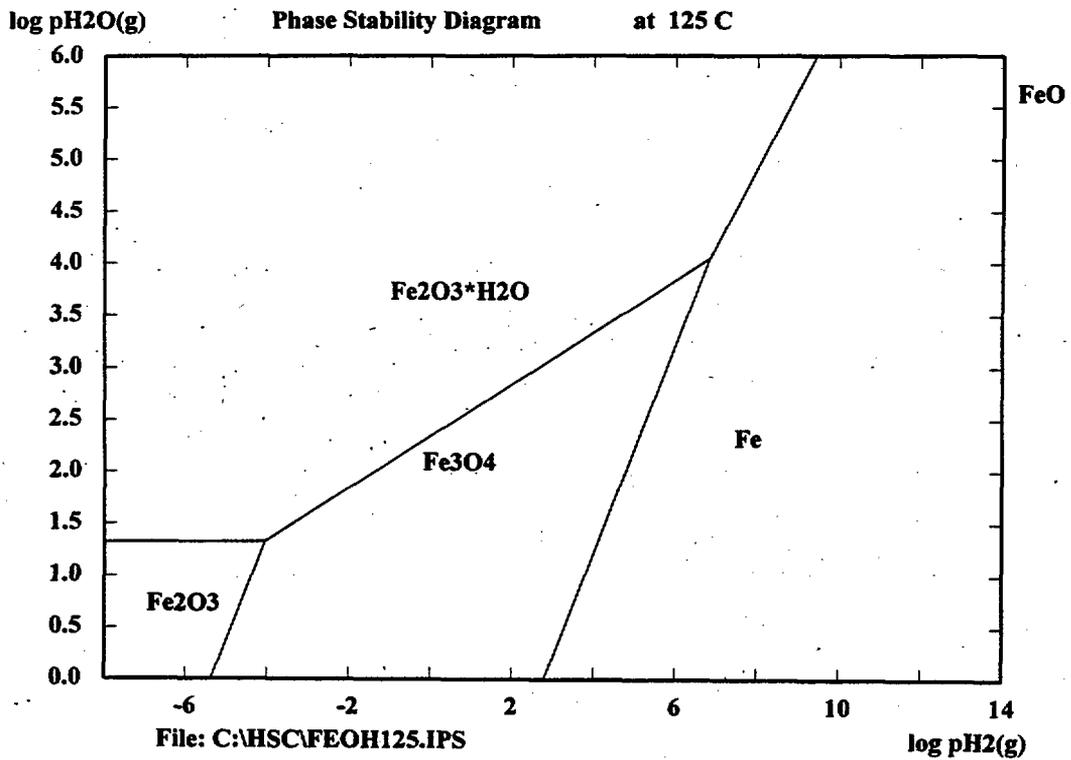


Figure 2.1-26 Phase stability for the iron oxide-water-hydrogen system at 125C.

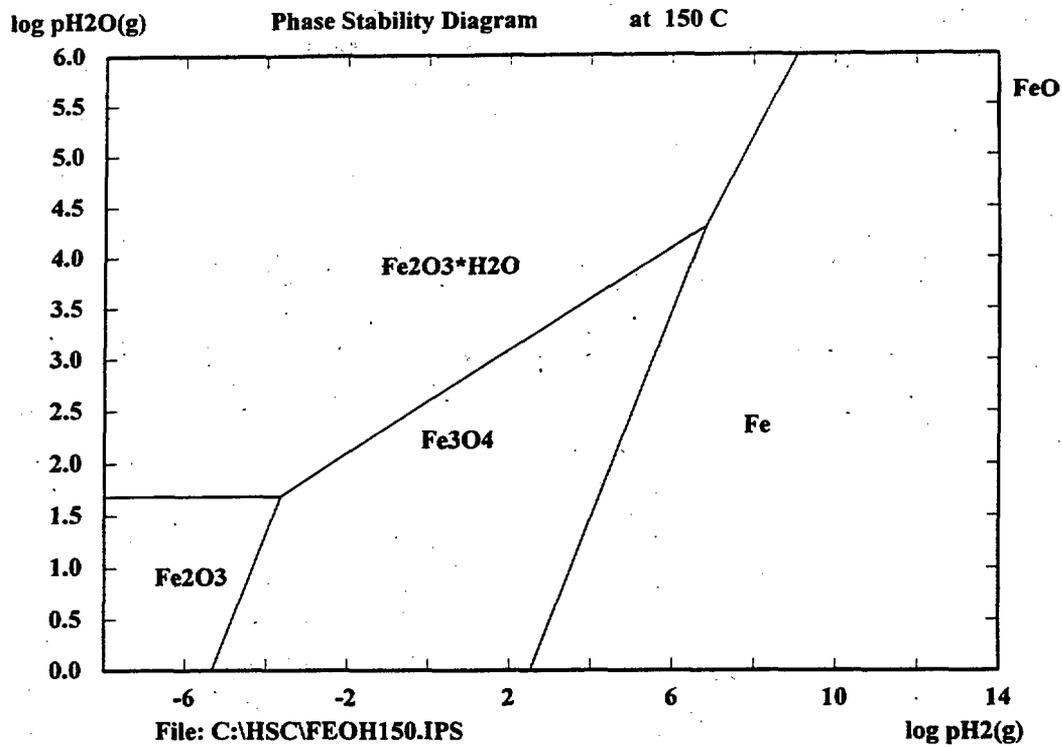


Figure 2.1-27 Phase stability for the iron oxide-water-hydrogen system at 150C.

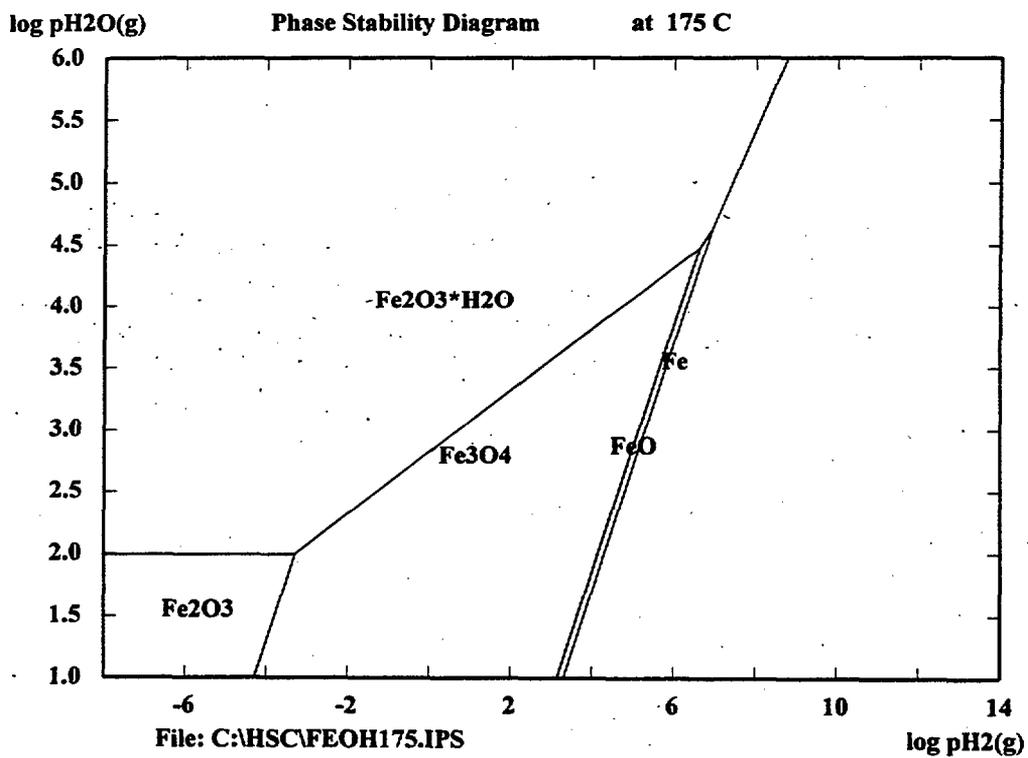


Figure 2.1-28 Phase stability for the iron oxide-water-hydrogen system at 175C.

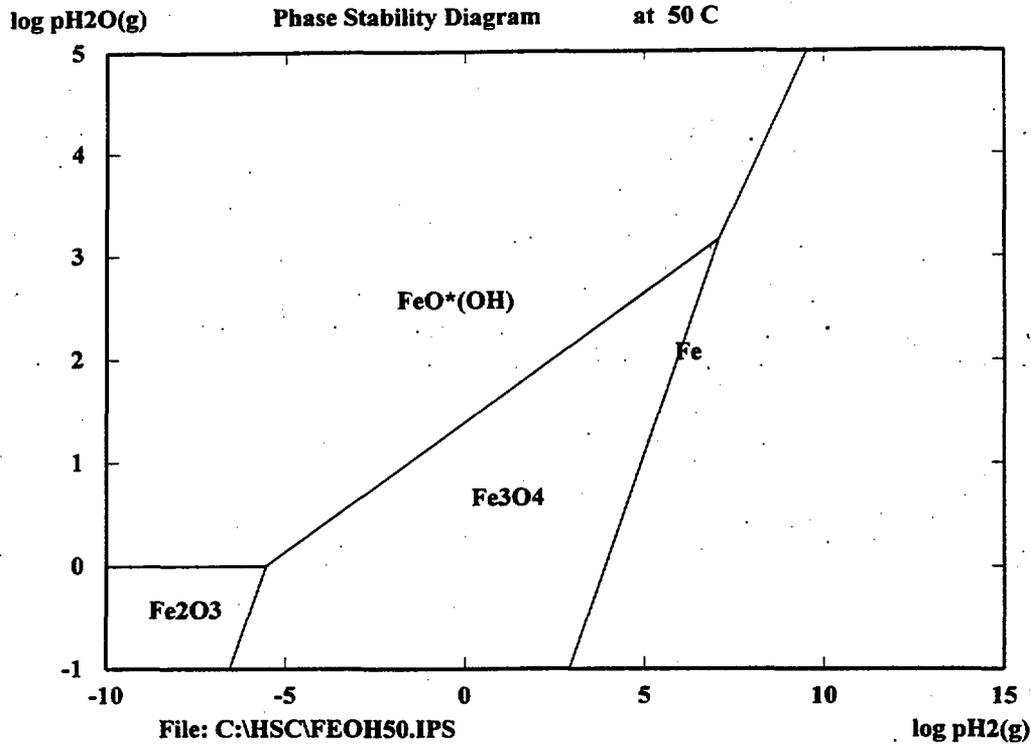


Figure 2.1-29 Phase stability for the iron oxide-water-hydrogen system at 50C.

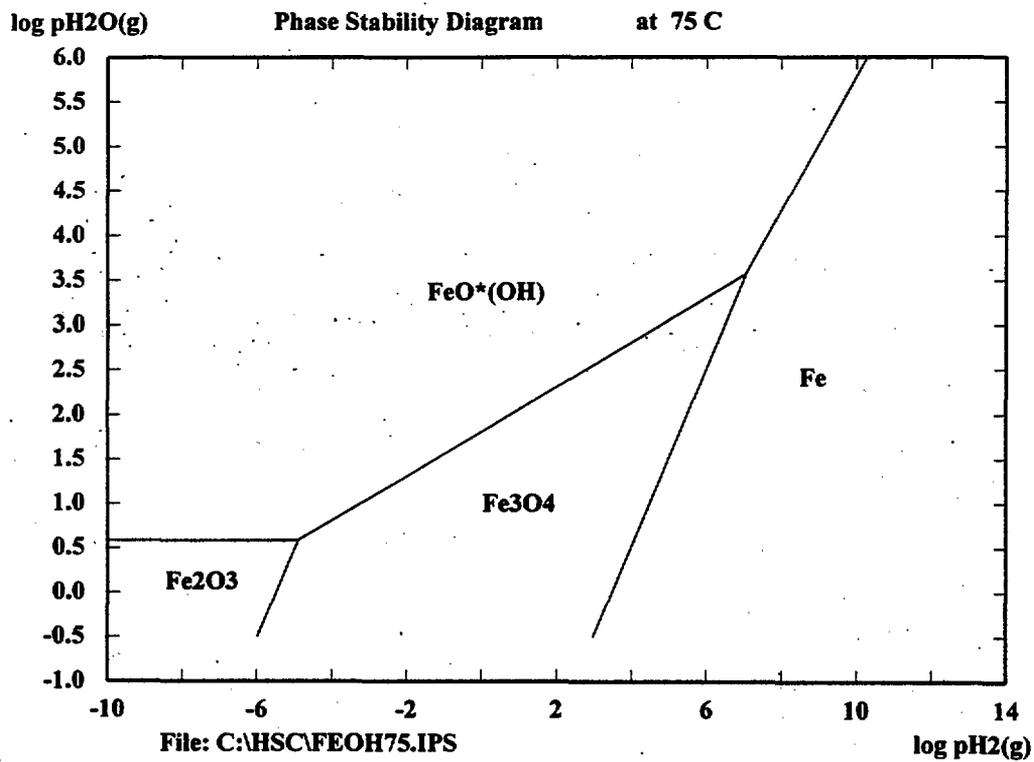


Figure 2.1-30 Phase stability for the iron oxide-water-hydrogen system at 75C.

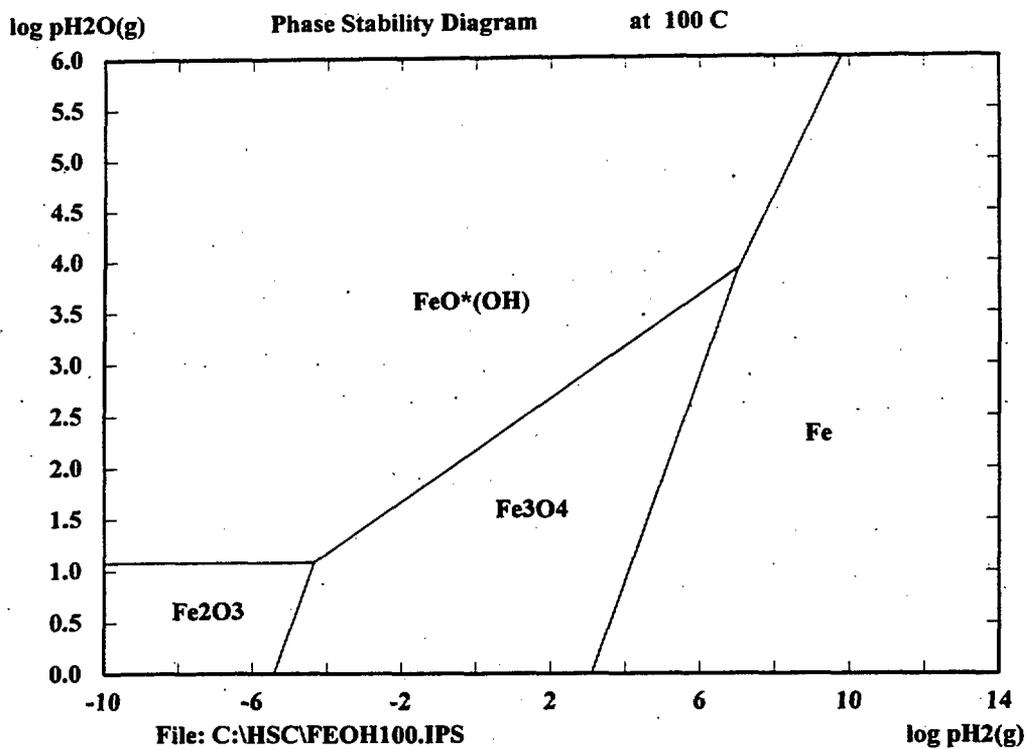


Figure 2.1-31 Phase stability for the iron oxide-water-hydrogen system at 100C.

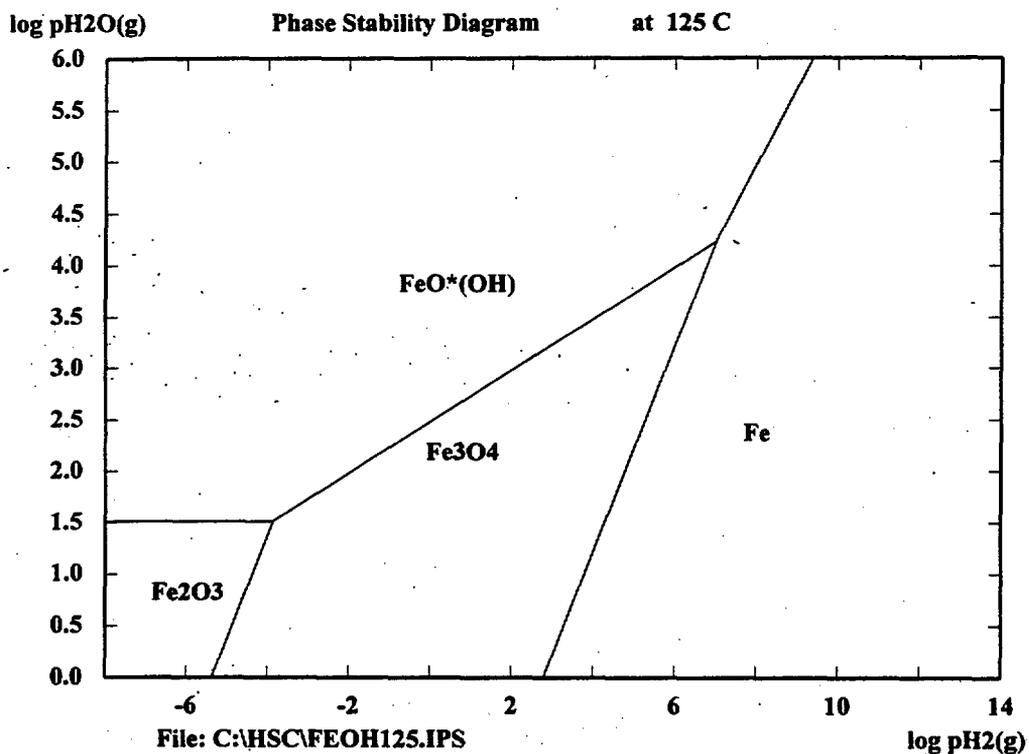


Figure 2.1-32 Phase stability for the iron oxide-water-hydrogen system at 125C.

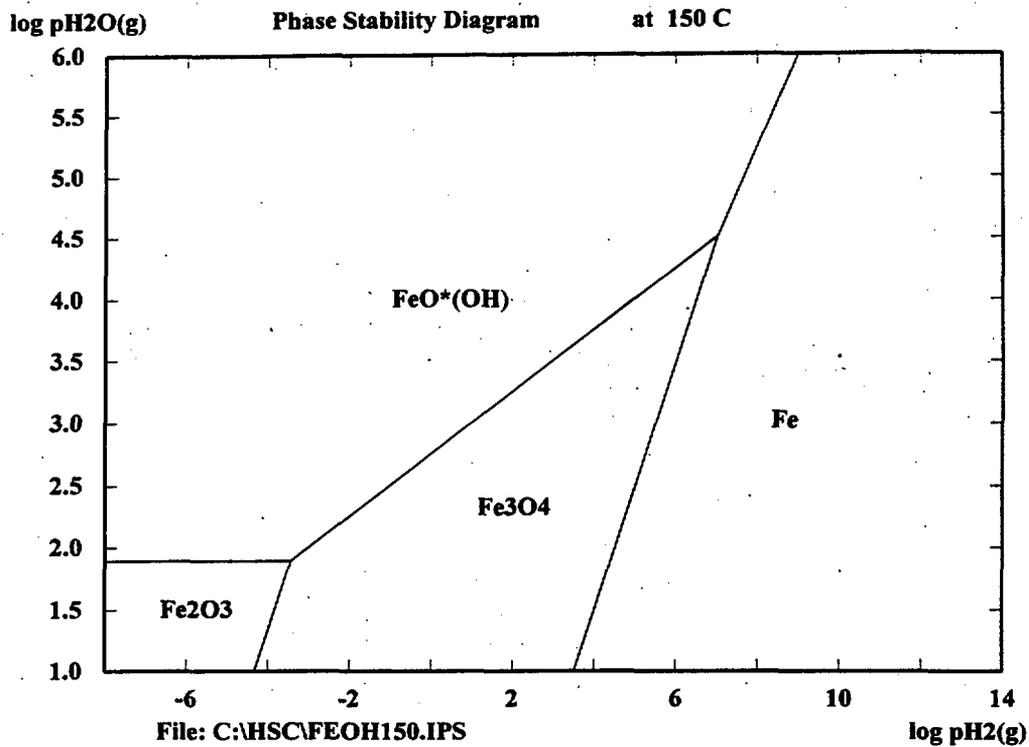


Figure 2.1-33 Phase stability for the iron oxide-water-hydrogen system at 150C.

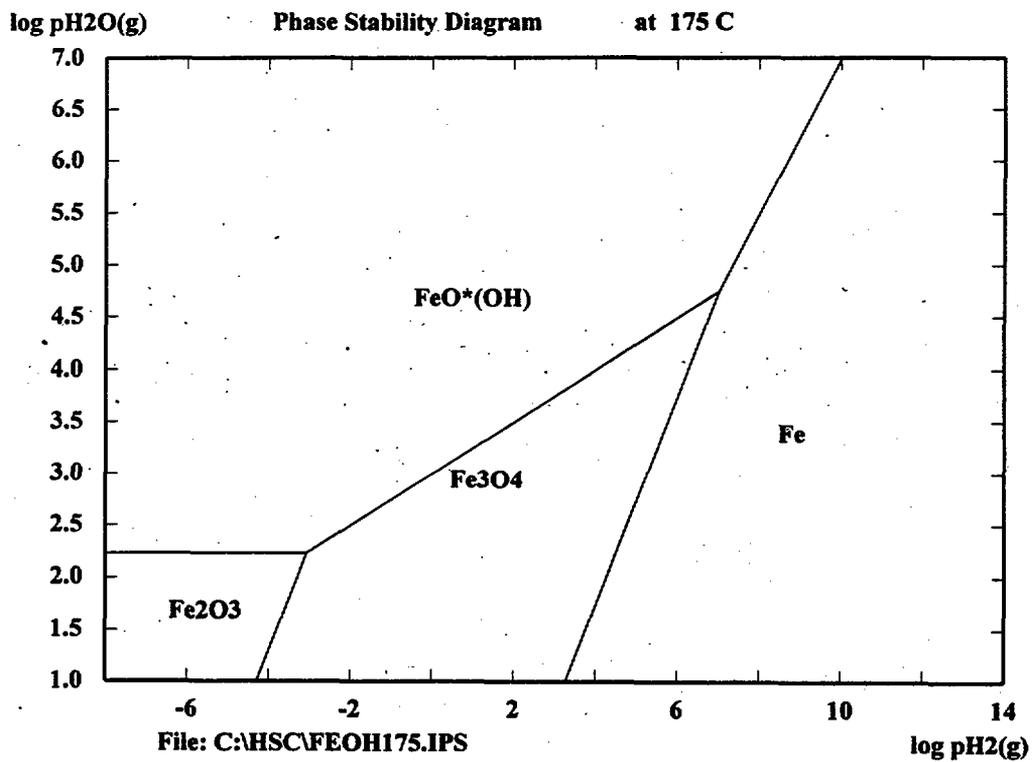


Figure 2.1-34 Phase stability for the iron oxide-water-hydrogen system at 175C.

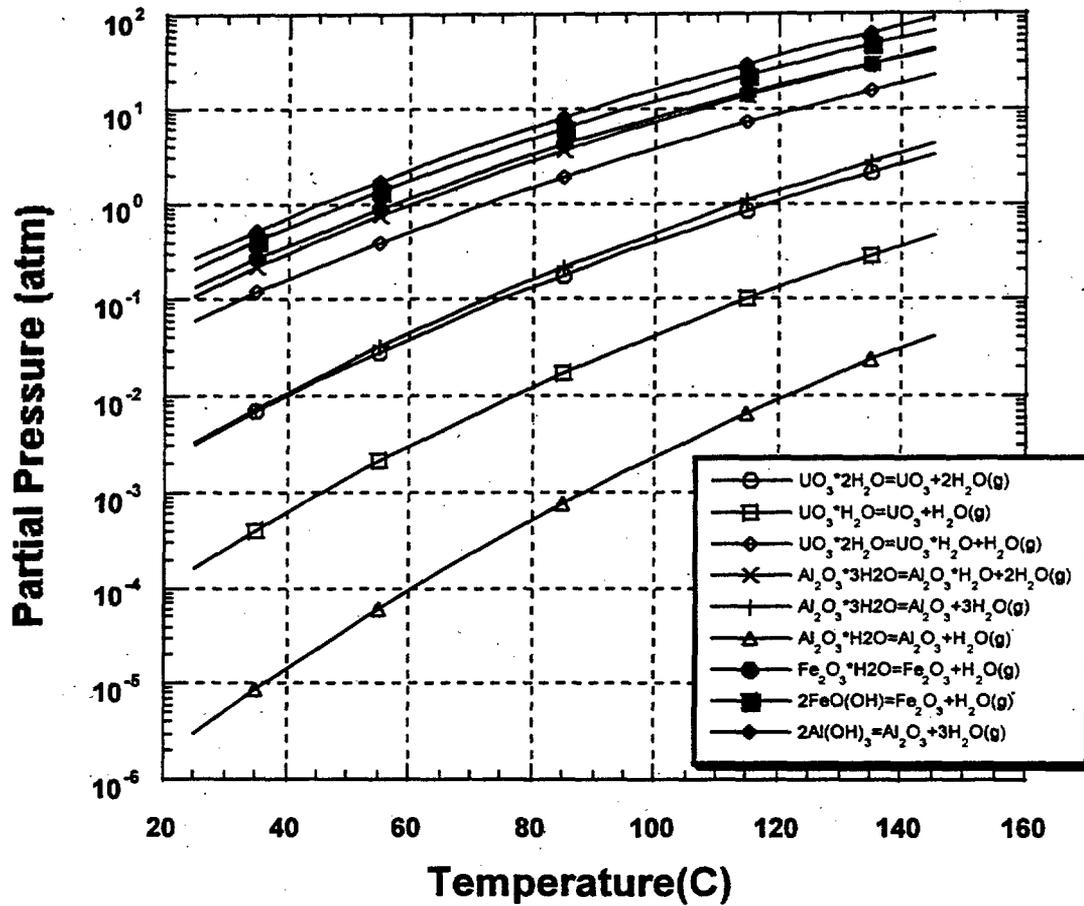


Figure 2.1-35 Equilibrium partial pressure of water over hydrated oxides or hydroxides as a function of temperature.

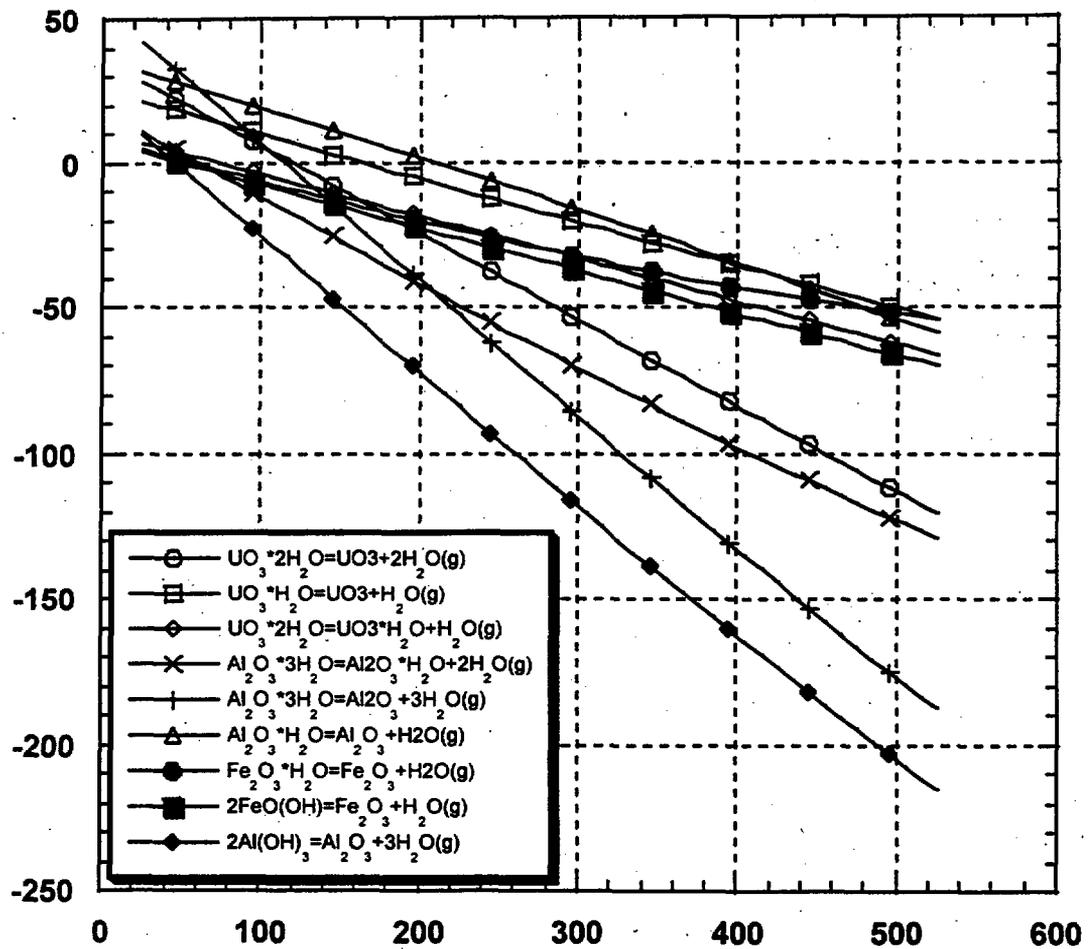


Figure 2.1-36 Free energy of decomposition for hydrated oxides and hydroxides as a function of temperature at 1 atmosphere pressure. A negative value for free energy means that the hydrated oxide or hydroxide is not stable.

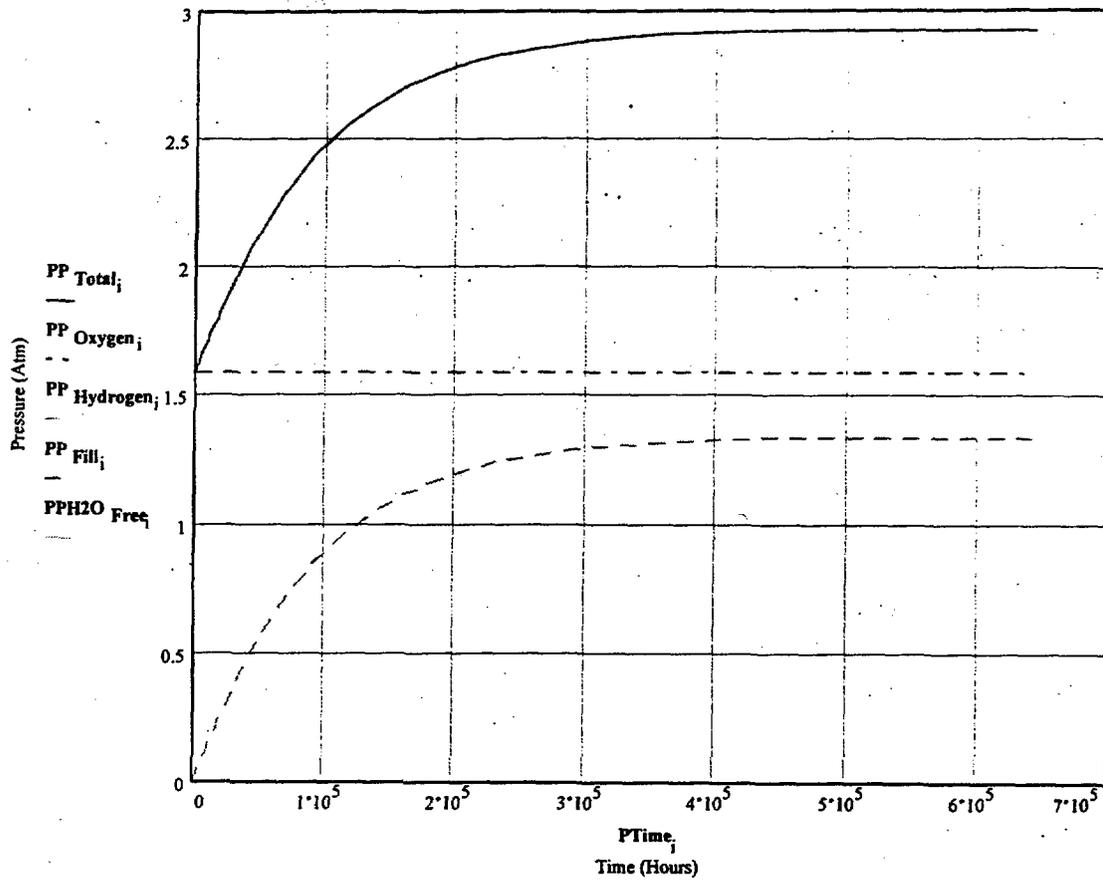


Figure 3.0-1. MCO Pressure Buildup for Base Case-5052 Aluminum

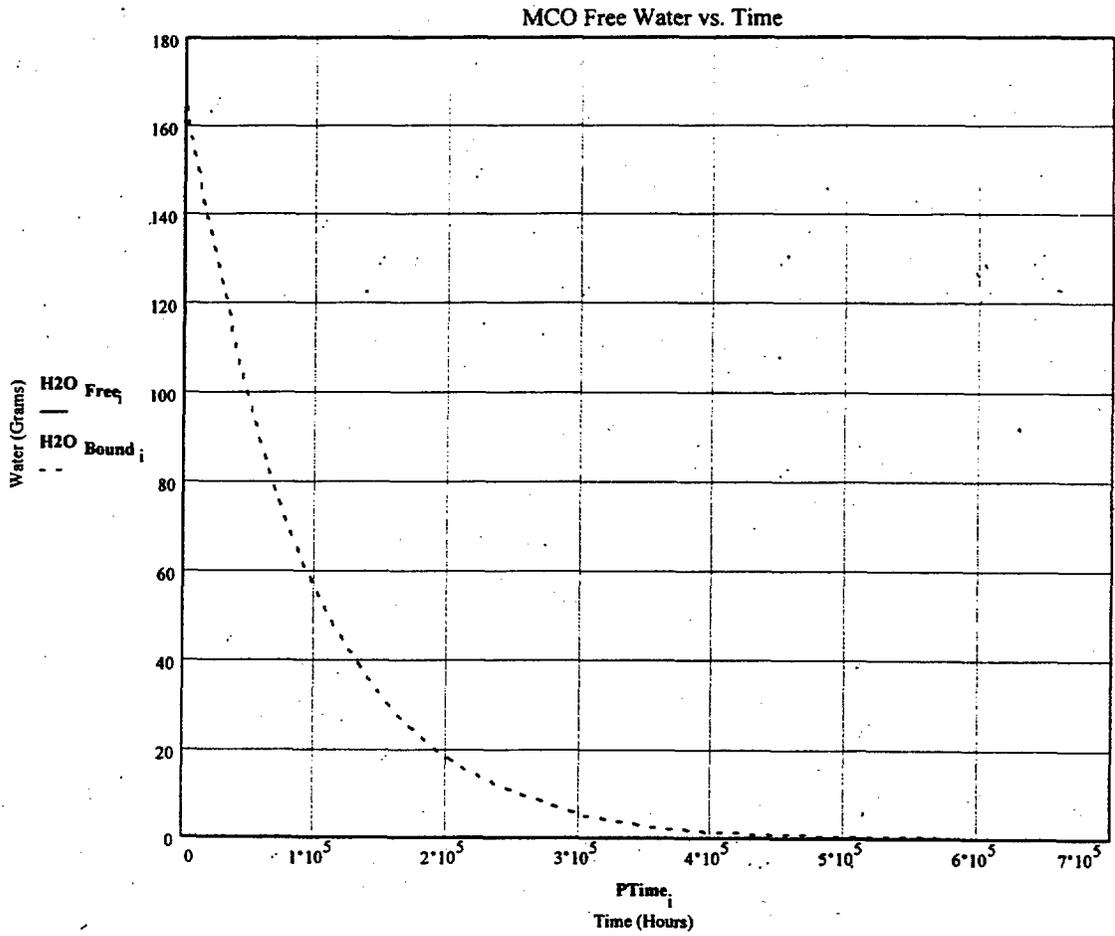


Figure 3.0-2. MCO Bound Water vs. Time for Base Case-5052 Aluminum.

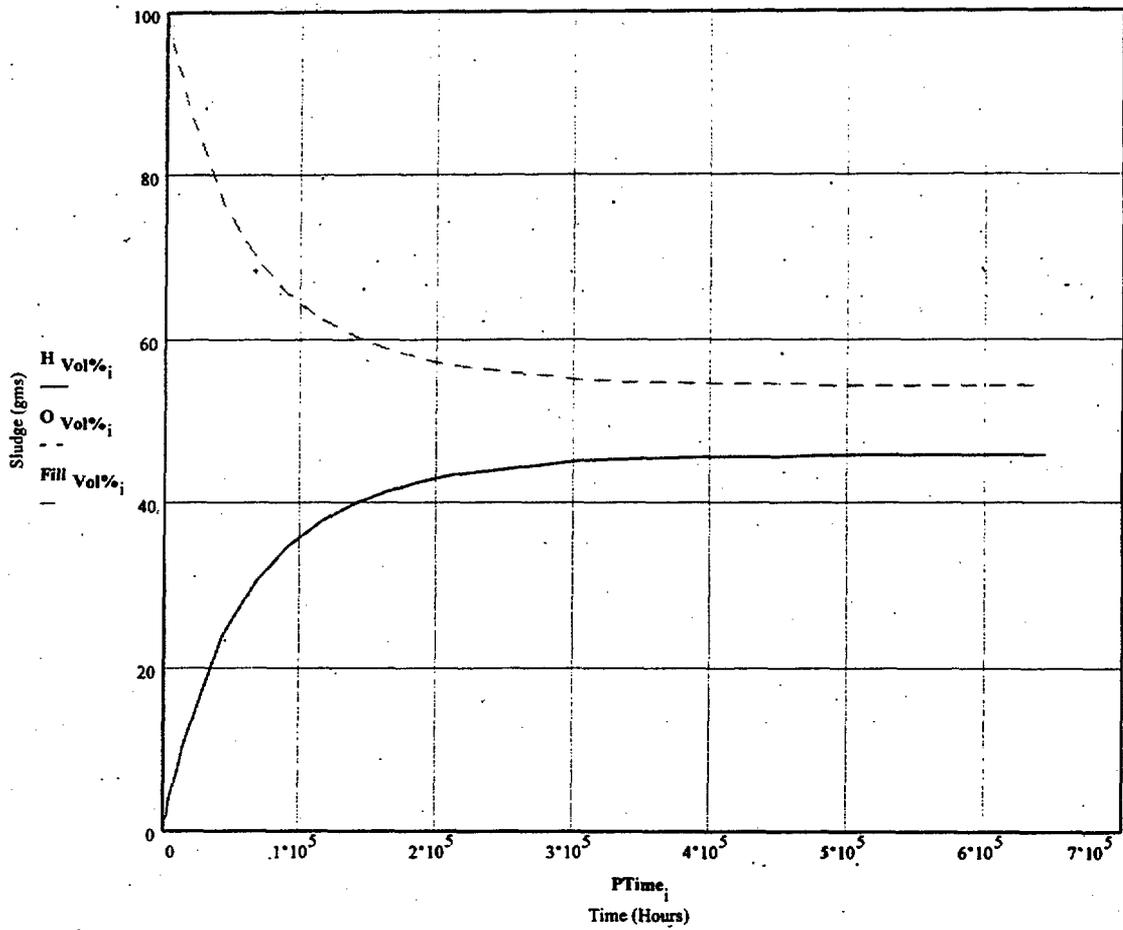


Figure 3.0-3. MCO Hydrogen/Oxygen vs. Time for Base Case-5052 Aluminum.

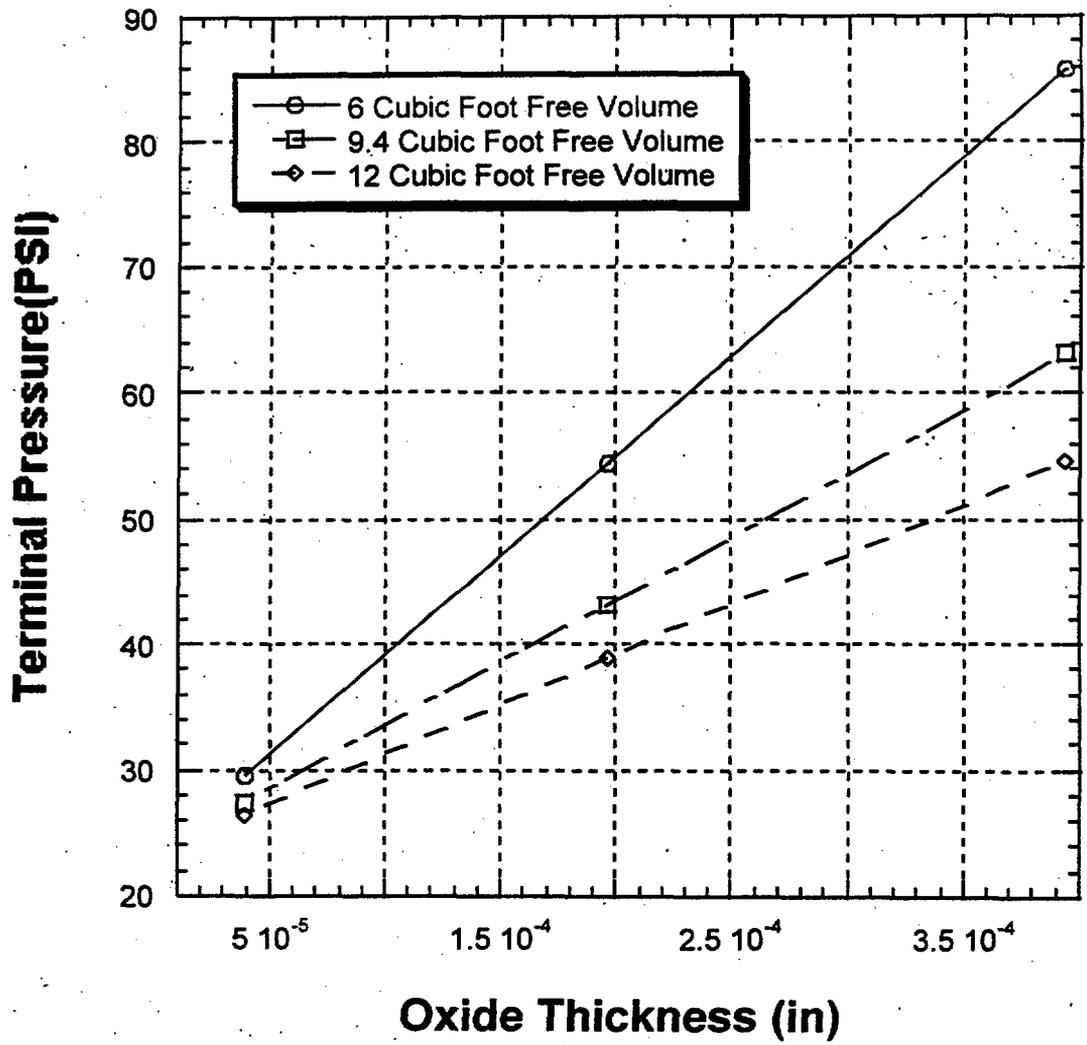


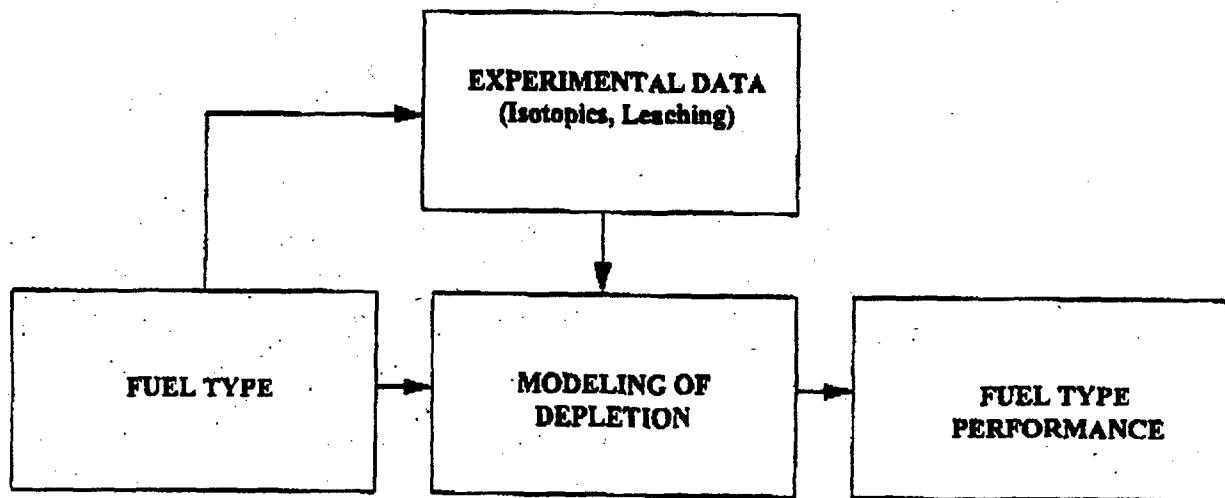
Figure 3.0.4. Terminal Pressure vs. Original Surface Oxide Thickness for the 32 Element Case for 5052 Aluminum

Attachment IID-7

Recommended Approach to SNF Characterization Requirements

Recommended SNF Characterization Requirements

The goal of the characterization program is to perform the minimum characterization necessary to satisfy repository criteria. Thus, the program would consist of a two level approach where actual characterization data is augmented by modeling results. Within each fuel type, a few elements would be analyzed (characterized) for the most significant characteristics with respect to repository requirements, (1) isotopic content and (2) leaching characteristics. Next, the expected performance of the fuel type will be estimated using neutronics/depletion analysis to calculate expected isotopic content in other elements. The experimental data from the selected elements will be used to calibrate the model results for each fuel type. This process is summarized as follows:



Appendix E
Performance Assessment of Research Reactor Spent Fuel in a Geologic Repository

Repository Evaluation Team:

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Hugh A. Benton, Framatome Cogema Fuels

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Appendix E
Performance Assessment of Research Reactor Spent Fuel in a Geologic Repository

E-1. Overview

The purpose of the performance assessment described herein is to determine the effects of disposal of aluminum-based research reactor fuel in a deep geologic repository. The aim of the preliminary assessment is to determine the incremental change in cumulative release and/or peak dose to an individual at the accessible environment that can be attributed to the disposal of the research reactor fuel.

E-2. Background on Performance Assessment

The U.S. Department of Energy Office of Civilian Radioactive Waste Management (DOE OCRWM) is currently investigating the feasibility of permanently disposing the nations' commercial high-level radioactive wastes (in the form of spent fuel from the over 100 electric power-generating nuclear reactors across the U.S.) and a portion of the defense high-level radioactive wastes (in the form of vitrified borosilicate glass which will be stored at federal facilities around the country) in the unsaturated tuffaceous rocks at Yucca Mountain, Nevada. Quantitative predictions based on the most current understanding of the process and parameters potentially affecting the long-term behavior of the disposal systems are used to assess the ability of the site and its associated engineered designs to meet regulatory objectives set forward by the U.S. Nuclear Regulatory Commission (NRC) and the U.S. Environmental Protection Agency (EPA).

The evaluation of the ability of the overall system to meet the performance objectives specified in the applicable regulatory standards has been termed total system performance assessment (TSPA). Total system performance assessments require the explicit quantification of the relevant processes and process interactions. In addition to providing a quantitative basis for evaluating the suitability of the site to meet regulatory objectives, such assessments are useful to help define the most significant processes, the information gaps and uncertainties regarding these processes and the corresponding parameters, and therefore the additional information required in order to have a more robust and defensible assessment of the overall performance.

E-3. Performance Assessment of Research Reactor Fuel

E-3.1 Performance Assessment Scenario

A bounding performance assessment of the Direct Co-Disposal option of research reactor fuel in a geologic repository was conducted based on one scenario from the Total System Performance Assessment-1995 (TSPA-1995).¹ The repository was assumed to be in the unsaturated zone about 300 meters above the ground-water table. For calculational purposes,

the repository inventory was assumed to be 63,000 MTU of commercial spent fuel, 7,000 MTU of vitrified defense high-level waste (DWPF-type product), and approximately 60 MTU of research reactor spent fuel. The assumption of a repository with a slightly larger than the Nuclear Waste Policy Act limit of 70,000 MTU was made for ease in calculation (i.e., the inventory used in TSPA-1995 did not have to be modified). The research reactor fuel is composed of approximately 25% high enriched and 75% low enriched spent fuel. For conservatism the research reactor spent fuel inventory was assumed to be composed entirely of the high enriched fuel. This assumption increases the inventory of ^{237}Np and actinides that decay to ^{237}Np which will increase the amount of release of ^{237}Np over the long-term (i.e., the largest dose over the long-term is from ^{237}Np and will increase because the calculated dose will be higher than if an average inventory was assumed). ^{237}Np was selected due to its maximum release rate that exceeds 0.1% of the NRC total-release-rate limit.

The research reactor fuel is assumed to be emplaced in the repository in 1,400 Direct Co-Disposal Waste Packages that contain five canisters of vitrified defense high-level waste around one canister of research reactor fuel assemblies which is centered in the package. The inventory of research reactor fuel per waste package is presented in Table 1, based on the assumptions that all of the fuel is highly enriched and that it is distributed equally among 1,400 packages. The waste package is assumed to have a 20 mm inner barrier of corrosion resistant material and a 100 mm outer barrier of corrosion allowance material. The waste packages are assumed to fail by a combination of general and pitting corrosion, initiated by temperature and relative humidity in the vicinity of the package.¹

The calculational scenario, selected from TSPA-1995, is for a repository thermal loading of 83 MTU/acre (the high thermal loading case from TSPA-1995), high infiltration rate (expected value of 1.25 mm/yr), and the presence of water dripping on some of the waste packages. Conceptually under this scenario, once the waste package has failed radionuclides diffuse out of the package, are transported through the engineered barrier system by diffusion or advection, and are transported through the unsaturated and saturated zones to the accessible environment. The accessible environment is 5,000 meters laterally from the edge of the repository. Thus, the transport path is about 300 meters through the unsaturated zone and 5,000 meters through the saturated zone. Along this transport pathway, processes such as sorption, dispersion, and dilution reduce the concentration of radionuclides.

The dissolution/alteration rate of the research reactor fuel was assumed to be the same as that of the commercial spent fuel. Because the dissolution/alteration rate for research reactor fuel is uncertain the results were also calculated for an assumed increase in the dissolution/alteration rate of a factor of ten.

Table E-1. Inventory of Research Reactor Spent Fuel² (assumes that the research reactor fuel is all highly enriched and emplaced in 1400 Direct Co-Disposal Waste Packages)

ISOTOPE	TOTAL CURIES (55.9 MTU)	CURIES PER PACKAGE
¹⁴ C	6.38×10^2	4.56×10^{-5}
⁷⁹ Se	2.30×10^2	1.64×10^{-1}
⁹³ Zr	1.18×10^2	8.43×10^{-1}
^{93m} Nb	8.80×10^2	6.29×10^{-1}
⁹⁹ Tc	7.74×10^3	5.53×10^0
¹⁰⁷ Pd	8.90×10^0	6.36×10^{-3}
¹²⁶ Sn	2.06×10^2	1.47×10^{-1}
¹²⁹ I	1.27×10^1	9.09×10^{-3}
¹³⁵ Cs	6.00×10^1	4.28×10^{-2}
¹⁵¹ Sm	1.65×10^5	1.18×10^2
²¹⁰ Pb	5.71×10^{-6}	4.08×10^{-9}
²²⁶ Ra	2.60×10^{-5}	1.86×10^{-8}
²²⁸ Ra	2.71×10^{-7}	1.94×10^{-10}
²²⁹ Th	4.83×10^{-5}	3.45×10^{-8}
²³⁰ Th	5.18×10^{-3}	3.70×10^{-6}
²³² Th	3.91×10^{-7}	2.79×10^{-10}
²³¹ Pa	4.09×10^{-2}	2.92×10^{-5}
²³³ U	2.73×10^{-2}	1.95×10^{-5}
²³⁴ U	3.14×10^1	2.25×10^{-2}
²³⁵ U	6.38×10^1	4.56×10^{-2}
²³⁶ U	2.62×10^2	1.87×10^{-1}
²³⁸ U	1.17×10^0	8.36×10^{-4}
²³⁷ Np	1.57×10^2	1.12×10^{-1}

Table E-1. Inventory of Research Reactor Spent Fuel² (assumes that the research reactor fuel is all highly enriched and emplaced in 1400 Direct Co-Disposal Waste Packages) (Continued)

ISOTOPE	TOTAL CURIES (55.9 MTU)	CURIES PER PACKAGE
²³⁸ Pu	5.35 x 10 ⁶	3.82 x 10 ³
²³⁹ Pu	1.67 x 10 ⁵	1.20 x 10 ²
²⁴⁰ Pu	9.54 x 10 ⁴	6.81 x 10 ¹
²⁴¹ Pu	8.11 x 10 ⁶	5.79 x 10 ³
²⁴² Pu	1.43 x 10 ²	1.02 x 10 ⁻¹
²⁴¹ Am	8.55 x 10 ⁵	6.11 x 10 ²
^{242m} Am	1.60 x 10 ²	1.14 x 10 ⁻¹
²⁴³ Am	5.92 x 10 ²	4.23 x 10 ⁻¹
²⁴⁴ Cm	8.07 x 10 ³	5.76 x 10 ⁰
²⁴⁵ Cm	1.25 x 10 ⁰	8.91 x 10 ⁻⁴
²⁴⁶ Cm	8.55 x 10 ⁻²	6.11 x 10 ⁻⁵
⁹⁴ Nb	1.28 x 10 ⁻²	9.12 x 10 ⁻⁶
²²⁷ Ac	1.47 x 10 ⁻²	1.05 x 10 ⁻⁵

E-3.2 Performance Assessment Results

Results were calculated for the expected annual dose to an individual over a one million year period after repository closure. Here "expected" is used in the sense of using the expected value of a distribution of values for various uncertain parameters. The word calculated is used throughout because over this long time period the results are highly uncertain and may not occur at all because of conservatism built into the performance assessment models. The calculated dose was assumed to be from drinking two liters of water per day from a well located at the accessible environment down gradient from the repository.

Figure E-1 shows the annual dose from the entire repository over one million years. The results for a repository without the research reactor fuel are not shown because when plotted on a log scale they appear to be identical to those shown in Figure E-1. The annual dose attributed to the research reactor fuel alone is shown in Figure E-2 and is more than three orders of magnitude below that from the entire repository. Because of the conservatism caused by assuming that all of the research reactor fuel is highly enriched, the results shown in Figure E-2 would be somewhat lower if the actual inventory of radionuclides had been used (i.e., by a factor of less than ten). Thus, Figure E-2 represents a worst case result for the scenario analyzed.

The results for the case in which the dissolution/alteration rate was increased by a factor of ten are not shown because they appear identical to the results shown in Figure E-2. The increased dissolution/alteration rate does not affect the height of the ^{237}Np peak which is controlled by the solubility of ^{237}Np and its parents.

The results of this performance assessment indicate that the dose from the research reactor spent fuel are more than three orders of magnitude below the doses from the entire repository and are masked by the 63,000 MTU of commercial spent fuel. The difference in dose with and without the 55.9 MTU of research reactor spent fuel are not discernable on a log plot of doses. Thus, the effect of adding research reactor spent fuel to the repository is considered to be insignificant. The long-term annual dose from the research reactor fuel is largely controlled by the inventory and solubility of ^{237}Np and its parents. Thus, disposal of other waste forms with similar inventories in a geologic repository should yield similar results.

1,000,000-yr Expected-Value Dose History 83/yes/hiq/clime/drips on WP

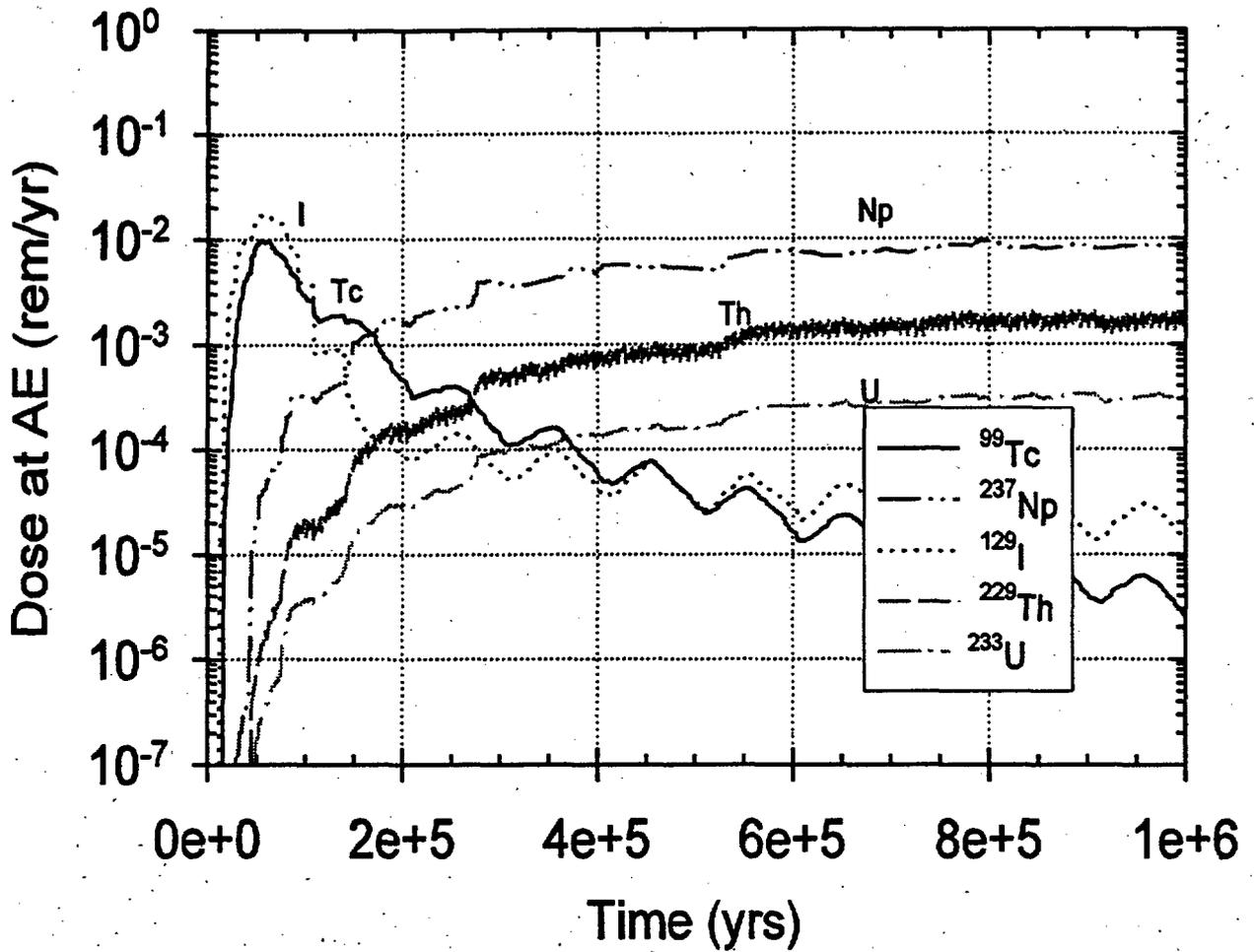


Figure E-1. Expected-Value Annual Dose History for One Million Years from the Entire Repository (TSPA-1995 Inventory Plus Research Reactor Fuel).

1,000,000-yr Expected-Value Dose History
83/yes/hiq/clime/drips on WP

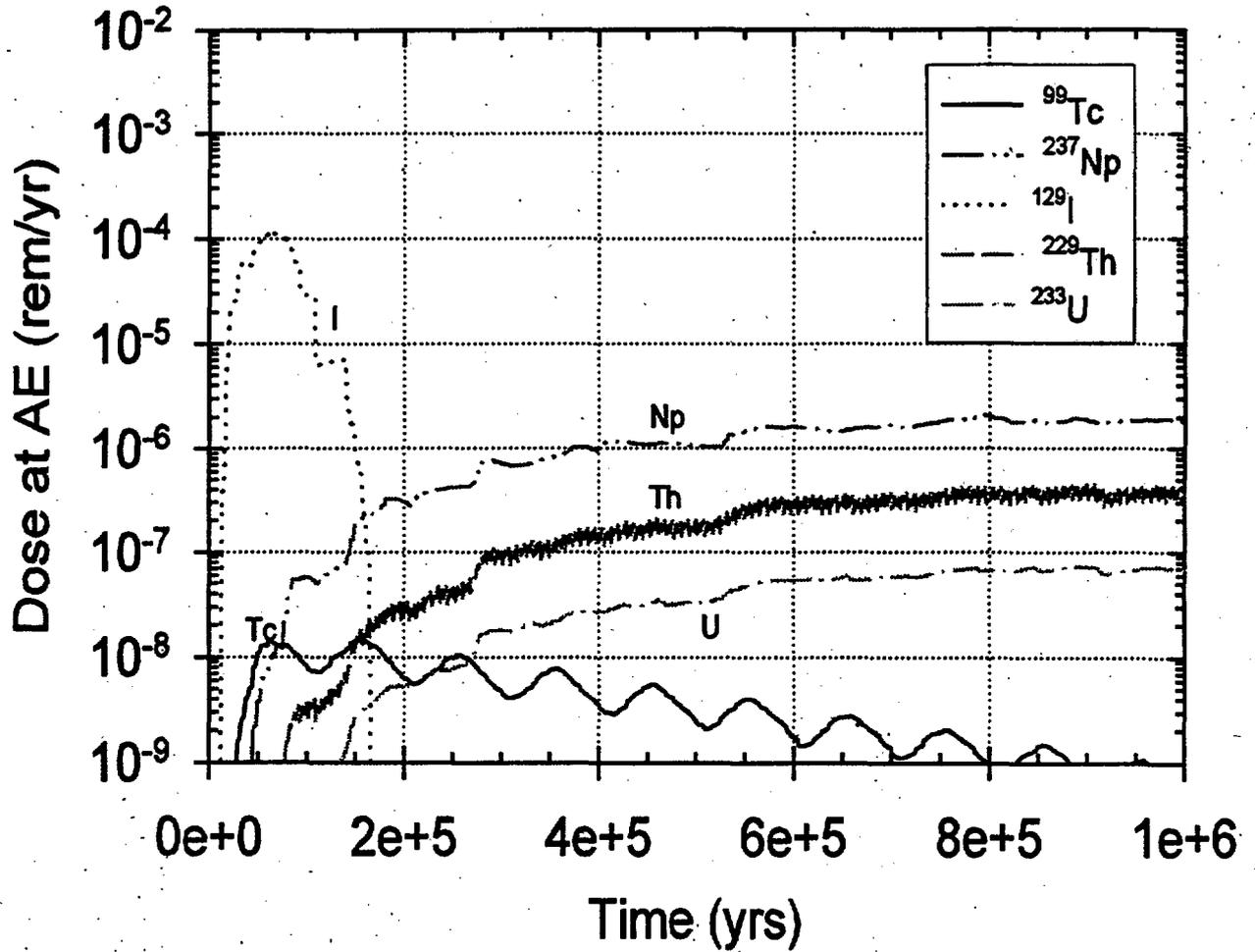


Figure E-2. Expected-Value Annual Dose History for One Million Years Attributed to the Research Reactor Fuel.

E-4. References

1. Andrews, R. W., J. E. Atkins, J. O. Duguid, B. E. Dunlap, J. E. Houseworth, L. R. Kennedy, J. H. Lee, S. Lingineni, J. A. McNeish, S. Mishra, M. Reeves, D. C. Sassani, S. D. Sevougian, F. Tsai, V. Vallikat, Q. L. Wang, and Y. Xiang, November, 1995. "Total System Performance Assessment - 1995: An Evaluation of the Potential Yucca Mountain Repository," B00000000-01717-2200-00136, Rev. 01, Civilian Radioactive Waste Management System, Management and Operating Contractor. Las Vegas, NV.
2. Research Reactor Fuel inventory modeled after 30-year cooled ATR high-enriched fuel utilizing ORIGEN. Data provided by fax from D. Cresap at the National Spent Nuclear Fuel Program on April 2, 1996.

Appendix F
Aluminum-Based Spent Nuclear Fuel Treatment and Disposal Technologies

Treatment Technology Evaluation Team:

J. Richard Murphy, Westinghouse Savannah River Company

Denzel L. Fillmore, Lockheed-Martin Idaho Technologies

Edward M. Burns, Westinghouse Savannah River Company

Appendix F
Aluminum-Based Spent Nuclear Fuel Treatment and Disposal Technologies

This appendix contains additional information regarding several of the aluminum-based spent nuclear fuel (SNF) treatment and disposal technologies evaluated by the Task Team. Technical papers have been included, where available, to provide detailed discussion of the processes and related research and development efforts. Summary information, similar to that shown on the Treatment Technology Synopses (Volume 1), is included where appropriate.

For each technology, at least one point of contact is provided as a source of further information. In addition, the majority of the technologies are discussed in the *DOE-Owned Spent Nuclear Fuel Technology Integration Plan* (DOE/SNF/PP-002, Revision 1, May 1996).

Note: Certain technical systems described in Appendix F were developed for applications other than SNF, such as the Glass Material Oxidation and Dissolution System (GMODS) description. Other treatment descriptions were developed for different packaging configurations, such as Press and Dilute.

The technical materials included here are as follows:

Press and Dilute

Includes Process Flow/Block Diagram

Melt and Dilute

Attachment: "Melt-Dilution - A Simplified Technology for Processing Aluminum Based Spent Nuclear Fuels."

Plasma Arc Treatment

Attachment: "A Direct, Single-Step Plasma Process for Converting Spent Nuclear Fuel and Associated Wastes into Vitreous Ceramic Final Waste Form: A Description."

Glass Material Oxidation and Dissolution System (GMODS)

Attachment: "Conversion of Plutonium Scrap and Residue to Borosilicate Glass Using the GMODS Process."

Dissolve and Vitrify

Attachment: "Vitrification of Spent Nuclear Fuel Executive Overview"

Electrometallurgical Treatment

Attachment: "Electrometallurgical Treatment of Aluminum-Matrix Fuels"

Chloride Volatility

Chop and Dilute/Poison

Can-in-Canister

Press and Dilute

Introduction

The Press and Dilute treatment system is a simple, mechanical volume compaction technique that is used to dilute the isotopic concentration of highly enriched uranium in aluminum-based fuels. The process discussed here involves the use of the 125-ton MPC; however, because other storage and transport alternatives are being considered, this part of the process and the diagram is not definitive as of yet.

Process

The Press and Dilute treatment process is as follows:

- a) The SNF is pressed into an approximately 1" thick by 3-1/2" wide by 35" long block.
- b) The pressed SNF is layered with depleted uranium (the U-238/U-235 ratio will depend on the enrichment desired).
- c) The depleted uranium and SNF are pressed to lock the pieces together, with dimensions of approximately 8" thick by 4" wide by 36" long.
- d) A total of 6 pressed blocks are placed into a stainless steel can approximately 8.8" OD square by 156" (maximum) long and up to 0.25" wall thickness.
- e) The can is welded shut.
- f) Finally, the can is placed into a storage/shipping container similar to the 125-ton MPC.

Waste Form

This pressed metal waste form will have the properties of solid aluminum, uranium and uranium-aluminum alloy. The waste form will be oxidized in the long term into metal oxides. By placing depleted uranium between the pressed highly-enriched uranium (HEU) or low-enriched uranium (LEU) SNF, the total package enrichment could be diluted below the commercial SNF limit of 2 wt% uranium-235. Thus, it is anticipated that the oxides of both uranium-235 and uranium-238 will be leached at a similar rate and thus both near and far field criticality potential, if any, will be similar to commercial SNF.

Advantages and Disadvantages

The primary advantages in this technique are controlled criticality (through use of the

depleted uranium), and the possibility of a reduced number of storage canisters (due to the "sandwiching" of waste blocks). The simple technology required to execute this process is also an advantage.

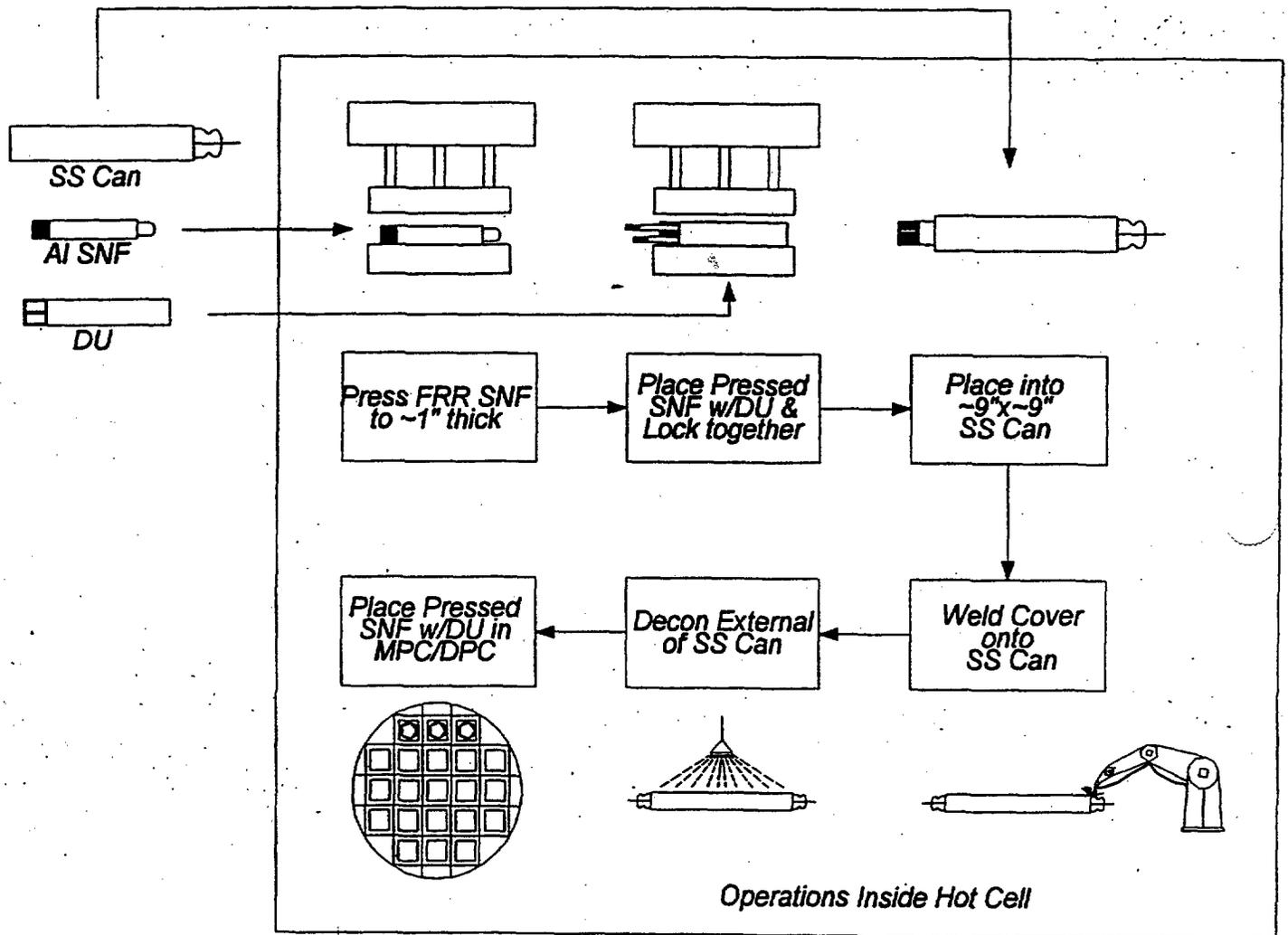
However, because of the dimensional requirements of the process itself, some SNF forms may not be suitable for this option. Fuel characterization must be known in order to process. Also, in order to prevent criticality in the repository, a low enrichment level is required for all waste material, and to keep the enrichment down a significant amount of depleted uranium must be used. This would require a larger number of repository waste canisters and a greater amount of repository space.

References

1. "Foreign Research Reactor Task Team Report - Press and Dilute/Poison Option," SNFP-FRR-Options-96-0001 Rev 0.

For Additional Information, Please Contact:

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Process Flow/Block Diagram
Press and Dilute/Poison Option

Melt and Dilute

Introduction

A melt-dilution process is being developed at Savannah River Site to provide a simplified, easily demonstrated technology for the conversion of aluminum-based spent nuclear fuels into a waste form suitable for emplacement in a geologic repository. This technology uses a single-step melting process to reduce the volume of the spent fuel, to reduce the uranium enrichment to prevent criticality risks, and to provide an easily characterized waste form while minimizing the creation of secondary wastes.

Process

- a) The fuel is cropped and melted with depleted uranium and/or nuclear poisons.
- b) The metallic waste form is compacted.
- c) The waste form is sealed in a corrosion resistant container which surrounds a multiphase aluminum alloy that incorporates such materials as fission products, uranium, and plutonium in an aluminum matrix.

Waste Form

If desired, by using slag-metal technologies, certain radionuclides may be selectively leached to produce a final metallic waste form containing few volatile fission product species. Utilization of this technology would convert the spent fuel into a waste form suitable for repository disposal with minimal impact to future disposition options should they be required. The melt-dilution process would provide a standard waste package that retains the uranium in a form suitable for recovery.

Advantages

1. Very low capital investment for the operation.
2. Superior waste form, and flexibility of the process to adaption for addition of poisons or glass frit.
3. Pretreatment of the fuel is not required.
4. Much of the procedural control and sensors technology can be adapted from the metals treatment industry.
5. Off-gas systems requirements are similar to existing systems.
6. The system is totally enclosed, eliminating radiation exposure hazards.
7. Characterization techniques can be adapted from the metallurgical industry.
8. The process can be tailored to specific types of spent nuclear fuel.
9. Secondary waste is minimal.

10. Other benefits such as meeting criticality requirements, reduced proliferation of fissile materials, and immobilization of waste are generic and comparable to the other treatment technologies.

Attachments

Attached is an additional writeup describing the Melt and Dilute treatment option.

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Melt- Dilution A Simplified Technology for Processing Aluminum Based Spent Nuclear Fuels

A melt-dilution process is being developed at SRS to provide a simplified, easily demonstrated technology for the conversion of aluminum based spent nuclear fuels into a waste form suitable for emplacement in a geological repository. The product of the melt reconfiguration treatment is a compacted (volume reduced), metallic waste form that contains uranium, fission products and plutonium in an aluminum matrix, all contained in a "container" which have corrosion characteristics superior to those used for the LWR waste forms. This technology uses a single step, melting process to reduce the volume of the spent fuel, reduce the uranium enrichment to minimize criticality risks and provide an easily characterized metallic waste form while minimizing the creation of secondary wastes. Further, if desired, by using slag-metal technologies, certain radio nuclides may be selective leached so as to produce the final metallic waste form containing little volatile fission products species.

Benefits of this process include very low capital investment for the operation, a superior waste form compared to the LWR waste forms, a flexible process which can be readily adapted to the addition of poisons, readily adaptable to the addition of glass frit to develop borosilicate glass waste form if necessary, low potential for airborne contamination and radiation exposure.

The process is based on the simultaneous melting of spent fuel sections in a permanent mold that is contained in a resistance furnace. The furnace is enclosed in a chamber which is connected to a off-gas system similar. Figure 1 shows a schematic of the furnace arrangement. The spent fuel gradually fills the container in the furnace. A lid or cap would be welded to the top of the mold to provide the waste form. Depleted uranium and/or other neutron poisons, if required, could also be incorporated into the melt during melt-dilution operation. Further, glass frit can also be added during the melt dilution process if borosilicate glass waste forms are desired for certain variety of spent fuels.

Laboratory scale melt-dilution/compaction facility capable of melting sections of Mark 22 aluminum clad spent nuclear fuel element, was developed at SRS, . A prototype furnace was also constructed. The melt-compaction process was demonstrated with non-irradiated aluminum clad fuel tubes. Full scale demonstrations, including cap welding and both destructive and non-destructive characterization and evaluation of the simulated waste form will follow the selection of an optimum container material. Demonstration tests using full length irradiated fuel can be subsequently initiated.

Although, the melt-dilution technology is primarily applicable to the aluminum based fuels, utilization of this technology would convert this category of DOE owned, spent nuclear fuels into a repository suitable waste form with minimal

impact to future disposition options should other options be required. The melt-dilution process will provide a standard waste form package that retains the uranium in form very suitable for recovery, should future generations desire to utilize the available energy. Additionally, if repository suitability is achieved simply through the introduction of nuclear poisons rather than dilution with depleted uranium, the highly enriched uranium could be recovered through a relatively simple metal smelting operation.

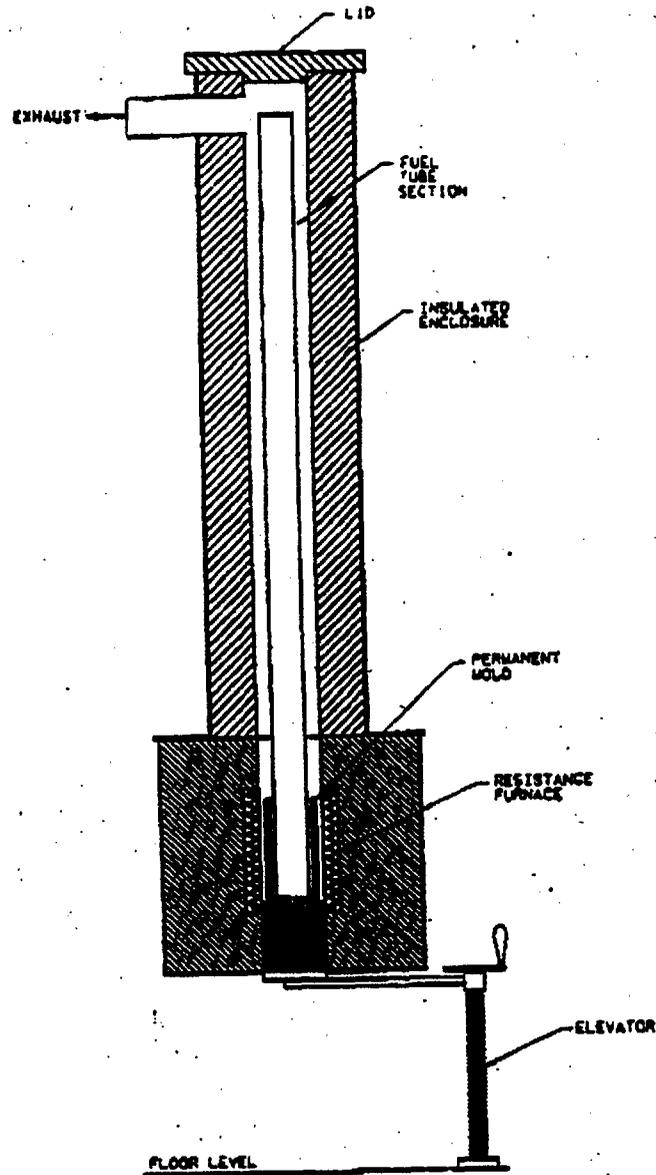


Figure 1: Schematic of Melt Dilution Furnace Facility

Plasma Arc Treatment

Introduction

The Plasma Arc Process is a feasible treatment plan that is primarily aimed at efficiently creating a safe waste form that has a high SNF loading capability. This single-step process requires little or no pretreatment, and therefore reduces worker radiation exposure risks and the separation costs of fuel decladding and size reduction.

The waste form is a vitreous ceramic material that is very durable due to its rigid crystalline lattice. It meets all of the conservative criteria required for the repository and is cost effective in its elimination of safety hazards. Proliferation risk is minimal due to the difficulty of recovering any fissile elements from this specific waste form.

Attachment

Attached is a document that explains in detail the process and product of this treatment option.

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A Direct, Single-Step Plasma Process for Converting Spent Nuclear Fuel and Associated Wastes into Vitreous Ceramic Final Waste Form: A Description

**X. Feng
R. E. Einziger
R. C. Eschenbach***

**Pacific Northwest National Laboratory
*Retech of Lockheed Martin**

January 1996 (Draft)

Summary

A single-step plasma arc process for converting aluminum-based highly enriched uranium (HEU) spent nuclear fuel (SNF) is proposed to be jointly developed by Pacific Northwest National Laboratory (PNNL) and Retech of Lockheed Martin (Retech). This technology is built on the extensive experience of nuclear waste form development and low-level nuclear waste treatment using the commercially available plasma arc centrifugal treatment (PACT) system through Retech. Spent nuclear fuel elements and associated wastes are loaded directly into a PACT-2.0 system with minimum characterization and pretreatment. An effective off-gas treatment system for plasma-arc process has been developed to treat volatile and semi-volatile materials for low-level/mixed nuclear waste treatment and a modification of the system is applicable for treating SNF, where there are far less volatile materials. The loaded filters and other wastes captured in the off-gas treatment are recycled back into the furnace to minimize secondary wastes. The vitreous ceramic waste form is similar to the borosilicate nuclear waste glass, except that it is more durable and has higher waste loading. The vitreous ceramic is suitable for permanent disposal in a geological repository and for interim storage. Criticality safety is assured through the use of "batch" mode (processing one batch at a time), and controlling the amount of fuel processed at one time. Minimum product volume is achieved because high-density vitreous ceramic is produced almost entirely from SNF and associated wastes with minimum additives and minimum secondary wastes. This plasma process is based on a mature commercial plasma-melting technology, extensive development for LLW nuclear wastes, and work between PNNL and Retech to develop a method of remote operation of plasma-arc processing for melting fuel cages in the late 1980s. It has the early involvement of a commercial partner. This plasma technology can be directly applied to treat many other types of DOE fuels besides aluminum highly enriched uranium SNF.

1.0 Introduction

A plasma-arc process (see Figure 1) for converting spent nuclear fuel (SNF) (e.g., aluminum, metallics, oxide, carbide, hydride, sodium-bearing, and zircalad) and associated wastes (sludge, soil, metal fragments, metal containers, and concrete debris) into a vitreous ceramic (VC) final waste form will be developed at Pacific Northwest National Laboratory* (PNNL) with support from Retech of Lockheed Martin (Retech), MSE, Inc.; Science Application International Company; Westinghouse Savannah River Company; and the University of New Mexico.

This process is based on plasma-melting technology developed and commercialized by Retech. The process combines the expertise and experience of PNNL, which developed and tested VC waste forms for the plasma-arc process and evaluated and integrated treatment technologies, and of the MSE and Retech, which developed and tested plasma arc centrifugal treatment (PACT) melting and conducted a large-scale demonstration of treating nuclear wastes using plasma technology.

Spent nuclear fuels in assemblies are loaded directly into the PACT furnace. A second feeder can add contaminated soils, sludges, concrete debris, and additives, simultaneously, if needed. Little or no pretreatment (characterization, sorting, size reduction, and separation) is needed for the processing, which results in reduced radiation exposure to personnel and lower treatment costs. The final product, a vitreous ceramic waste form, is produced at the end of each batch of melting.

2.0 Description

2.1 Minimum SNF Characterization

The vitreous ceramic waste form has the flexibility to accommodate feed composition changes within large ranges; the elemental composition information needed for waste-form formulation usually can be obtained through information on the types and the history of fuel production and service. If such information is not available, then rough analyses through X-ray diffraction or other types of inexpensive methods to provide elemental compositions with ± 5 wt% accuracies are sufficient. If the service history cannot provide highly enriched uranium (HEU) and actinide contents accurate enough for criticality control, then an adequate analysis of these elements will be required; similar to other treatment technology. There is no need to analyze for accurate contents of hydrides, water, organics, and surface areas, and to assess the conditions of the fuels, as is required by some other technologies.

2.2 Minimum Pretreatment

Minimum pretreatment is required because SNF in assemblies can be loaded directly (a required element of every furnace or any other technology) into the PACT system feeder without size reduction, clad and fuel separation, or any other pretreatment. The fuel assembly is sliced into small pieces by a plasma torch before being dropped into the centrifugal tub of the furnace. Other associated wastes, such as sludge, concrete debris, metal fragments, fuel containers, sand, soil, other basin debris, and ion-exchange materials, also can be fed directly into the furnace without sorting, separation, size reduction, and drying.

2.3 A One-Step Process

A schematic of the plasma process is provided in Figure 2.1. The major components of the PACT system are the feeder; the primary chamber with rotating tub; the plasma torch; an SCC with after-burner (which is optional); a vitreous ceramic collection chamber; and an off-gas treatment system.

From the time a fuel assembly is fed into the rotating tub (together with other associated wastes or additives, if necessary) from the side of the furnace, until the vitreous ceramics are discharged from the bottom of the furnace, there are no discrete individual steps that can be distinguished; the process can be stopped at an intermediate step without harm to the process as a whole. There is no material transfer from one tub to another tub (except for the off-gas) and no evolution-of-use of different instrumentation during the process. Therefore, it is fair to say that this is a single-step process, from feeding to end product. It is, however, possible to divide the melting process conceptually into different stages, as described below.

Feeding Stage: The feeding SNF assembly is sliced and dropped into the rotating tub as the SNF element is fed into the furnace;

- **Melting and Oxidizing Stage:** The fed materials are melted and oxidized under the plasma torch;
- **Melting Stage:** The melted and oxidized materials are homogenized during centrifuging and under the depression of the torch force; and
- **Discharging Stage:** The rotation speed of the rotating tub slows down and vitreous ceramic is discharged into a canister.

All of the above stages are continuous in a single-step process.

2.4 Off-Gas Treatment

The plasma-arc process is similar to any other high-temperature process in which glasses or ceramic waste forms are produced, in that some particulates and volatile and/or semi-volatile elements will escape from the melts. An off-gas treatment system is necessary to capture these particulates and volatile/semi-volatile elements. Extensive work has been performed during the development of the plasma-arc process for treating mixed and low-level wastes where many more volatile elements are presented (MSE, Inc. 1994; Hassel 1994). These developed off-gas systems have been shown to be in full compliance with Environmental Protection Agency emission requirements. The used filters (baghouse, glass filter, or HEPA filters) are recycled back to the furnace to be made into vitreous ceramic, to minimize secondary waste generation. The existing offgas systems can be modified as necessary for treating SNF.

2.5 Criticality Control

Criticality safety is ensured through equipment design and operation in batch mode. A PACT-2.0 system will be used as a full-scale system during SNF treatment. This system has a rotating tub of 2.0 feet or less in diameter (the geometric limit will be determined through criticality considerations); only critically safe amounts of HEU and Pu are allowed in each batch of melting. The melted vitreous ceramics are completely discharged and the rotating tub is checked for possible HEU accumulation before the next batch of a melt begins. Criticality can occur with HEU metal in sufficient quantities, but it cannot occur when the HEU is dissolved in the vitreous ceramics at an appropriate concentration, such as below 4% while the total uranium content can be as high as 40 wt%. The incorporation of neutron poisons such as Gd or Hf in the vitreous ceramics can be used to further reduce the possibility of nuclear criticality. Gd and Hf have been shown to have chemical behavior similar to uranium in vitreous ceramics, and they occupy the same lattice spaces as uranium and plutonium.

2.6 Waste Streams

Spent nuclear fuel and associated wastes are made into vitreous ceramic final waste form. Secondary wastes from off-gas treatment and the used filters are recycled to the furnace and immobilized into vitreous ceramics. A limited amount of water used in the off-gas treatment may be treated through the existing wastewater treatment facility at the DOE site; the loaded ion-exchange materials can

also be made into vitreous ceramics. A limited amount of fission gas is generated, as would be the case with many other proposed technology where glasses or ceramics are produced which involves high processing temperatures.

Vitreous ceramics promote the formation of stable and low-solubility (in water) crystalline phases that are embedded in a glass matrix rich in network formers such SiO_2 and Al_2O_3 . The crystals formed (e.g., zirconolite, perovskite, zircon, and spinel) can (1) incorporate large contents of uranium (more than 30 wt%), plutonium (about 15 wt%), other fission products, and neutron poisons such as Gd, Hf, and other rare earths in the lattices; (2) incorporate a large amount of Al, Fe, Ni, Cr, Zr, and Ti oxides to accommodate the large metallic content of the SNF and the associated sludge; (3) use little or no network-forming elements, such as silicon, so these latter elements are enriched in the residual glassy matrix; (4) be durable and insoluble in water because these crystals duplicate those which have existed in nature for billions of years; and (5) tightly bind to the glassy matrix so the physical integrity and mechanical strength of the waste form can be maintained. The residual glassy matrixes in vitreous ceramics are enriched in network formers and deficient in alkalis and are therefore also more durable than high-level waste glasses. Furthermore, the residual glassy matrix can immobilize any hazardous elements that cannot be incorporated in the crystals and offers much higher flexibility than pure crystalline waste forms, such as synroc and zircon. The targeted crystals to be formed in each melt depend on the types of fuel and associated wastes to be fed. The loading of aluminum SNF in vitreous ceramics is smaller than that for zircaloy SNF, but is much higher than that in a glass.

Recent work (MSE, Inc. 1994; Hassel 1994; Feng 1994; Feng et al. 1994a; Feng et al. 1994b; Feng et al. 1994c) indicates that vitreous ceramic final waste forms are similar to borosilicate glass waste forms, but are more durable and can have higher waste loading than a glass. Vitreous ceramic waste forms have been increasingly recognized as a viable final waste form by the waste management efforts both within and outside of the United States. This waste form has been used by the U.S. Department of Energy's Minimum Additive Waste Stabilization Program (Feng et al. 1994a; DOE 1994) for mixed waste; by WINCO of Idaho Falls for calcined high-level wastes (Vinjamuri 1994); by MSE, Inc. (Rattleson 1994); and SAIC (Geimer et al. 1994); for Buried Waste Integrated Demonstration, by EG&G (Quapp 1993); for spent fuel, by Switzerland (Hoffelner et al. 1992); for low- and intermediate-level waste, by Russia (Dmitriyev 1994; Epel'baum et al. 1994; Sobel et al. 1994); for various wastes, by Australia (White 1994); and by other international efforts (Lutze and Ewing, 1988).

4.0 References

- Battleson, D. 1994. "Latest Developments of Plasma Technology at the CDIF," *Proc. Spectrum '94*, Atlanta, Georgia.
- Dmitriyev, S.A., S.V. Stefanovsky, I.A. Knyazev, F.A. Lifanov. 1994. "Characterization of Slag Product from Plasma Furnace for Unsorted Solid Radioactive Waste Treatment," *Sci. Basis Nucl. Waste Manag.*, Philadelphia, Pennsylvania.
- Epelbaum, M.B., A.S. Chekhir, T.P. Salova, A.I. Golodnova. 1994. "Matrices From Magmatic Rocks for Immobilization of Radioactive Waste," *Sci. Bas. Nucl. Waste Manag.*, Philadelphia, Pennsylvania.
- Feng, X. 1994. "Development of Vitreous Ceramic as Final Waste Forms for Mixed Wastes," *Proc. American Chemical Society "Emerging Technologies in Hazardous Waste Management VI,"* September 19-21, 1994, Atlanta, Georgia.
- Feng, X., G. Ordaz, and P. Krumrine. 1994a. "Glassy Slag - A Complementary Waste Form to Homogenous Glass for the Implementation of MAWS in Treating DOE Low-level/Mixed Wastes," *Proc. Spectrum '94*, August 14-18, 1994, Atlanta, Georgia.
- Feng, X., D. J. Wronkiewicz, J. K. Bates, N. R. Brown, E. C. Buck, N. L. Dietz, M. Gong, and J. W. Emery. 1994b. "Vitreous ceramic For Minimum Additive Waste Stabilization, Interim Program Report, May 1993 - February 1994," Argonne National Laboratory Report # ANL-94/24, Argonne, Illinois.
- Feng, X., D. Wronkiewicz, N. Brown, M. Gong, C. Whitworth, K. Filius, and D. Battleson. 1994c. "Comparison of Glassy Slag Waste Forms produced in Laboratory Crucibles and in a Pilot Scale Plasma Furnace," *Proc. American Chemical Society special symposium "Emerging Technologies in Hazardous Waste Management VI,"* September 19-21, 1994, Atlanta, Georgia.
- Geimer, R., C. Dwight, and G. McClellan. 1994. "The Plasma Hearth Process Demonstration Project For Mixed Waste Treatment," *Pro. Spectrum'94*, Atlanta, Georgia.
- Hassel, G. R., J. A. Batdorf, R.M. Geimer, G.L. Leatherman, J.M. Wilson, W.P. Wolf, A. Wollerman. 1991. "Evaluation of the test results from the plasma hearth process," Science Applications International Corporation, Report #SAIC-94/1095, Idaho Falls, Idaho.
- Hoffelner, W., A. Chrubasik, R.C. Eschenbach, M.R. Funtschilling, and B. Pellaud. 1992. "Plasma Technology for Rapid Oxidation, Melting and Vitrification of Low/Medium Radioactive Waste," *Nuclear Eng. Intern.*,
- Lutze W., and R.C. Ewing. 1988. *Radioactive Waste Forms for the Future*, North-Holland, Amsterdam.
- MSE, Inc. 1994. "Test results from plasma centrifugal furnace demonstrations under the buried waste integrated demonstration," Report #PCF-D026, Butte, Montana.
- Quapp, W.J., T.L. Eddy, G.A. Reimann, R.L. Miller. 1993. "Feasibility of Vitrification of Spent Nuclear Fuel Using Iron-Enriched Basalt," *Proc. Emerging Tech. Hazardous Waste*

Manag., Washington, DC.

Sobolev, I., S. Stefanovsky, F. Lifanov. 1994. "Synthetic Melted Rock-Type Waste Forms," *Sci. Bas. Nucl. Manag.*, Philadelphia, Pennsylvania.

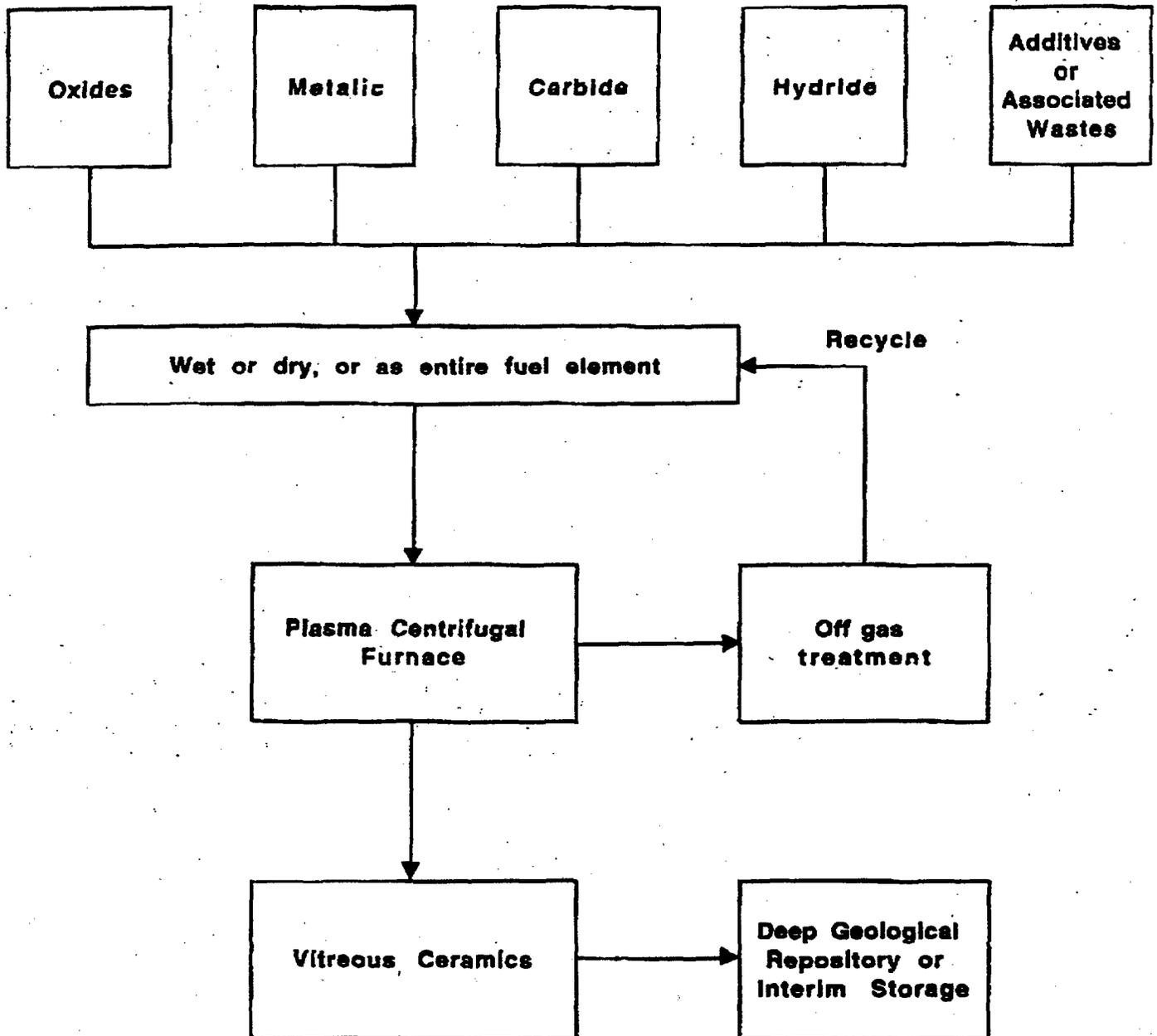
U. S. Department of Energy, Office of Technology Development. 1994. "Minimum Additive Waste Stabilization (MAWS)," Technology Summary DOE/EM-124P. Washington, DC.

Vinjamuri, K. 1994. "Solid-Based Glass-Ceramic Waste Forms for Immobilization of the Fluorine/Sodium Calcined High-Level Waste Stored at the Idaho Chemical Plant," *Proc. Spectrum'94*, Atlanta, Georgia.

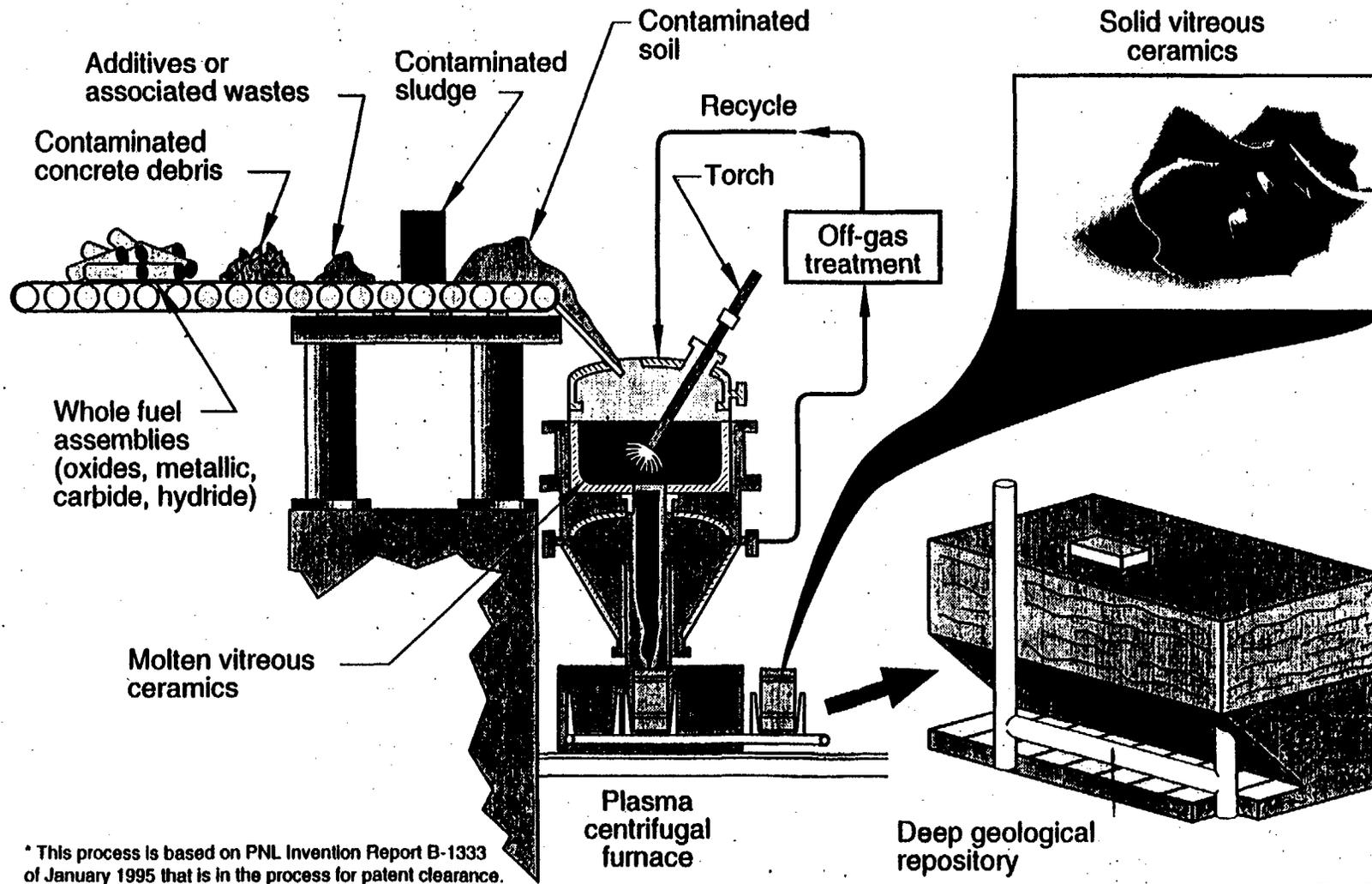
White, T.J. 1994. "Synthetic Mineral Immobilization - A Revitalization Technology," *Sci. Bas. Nucl. Waste Manag.*, Philadelphia, Pennsylvania.

A Direct and Single-Step Plasma Arc Process for Spent Nuclear Fuel

— PNL



A Block Diagram of the Plasma Arc - Vitreous Ceramic Process for Treating Spent Nuclear Fuels



* This process is based on PNL Invention Report B-1333 of January 1995 that is in the process for patent clearance.

R95 1128

The Plasma Arc Process.

Glass Material Oxidation and Dissolution System (GMODS)

Introduction

GMODS is a treatment technology for the direct conversion of metals, ceramics, and amorphous solids to borosilicate glass. Organics are oxidized with the residues converted to glass. Chlorides are converted into a low-chloride borosilicate glass and secondary sodium chloride stream. This capability allows direct conversion of SNF to glass. GMODS is a new glass making treatment technology, using unusual starting materials, but does not produce a new glass waste form.

Advantages

The GMODS process is a general purpose process for conversion of radioactive waste to glass. If GMODS was developed to convert aluminum-based spent fuel to HLW glass, the option exists to later use the facility to convert other highly radioactive solid wastes to glass. Such wastes could include (1) other miscellaneous spent nuclear fuels, (2) hot cell wastes, (3) remote handled transuranic wastes, and (4) highly radioactive decommissioning wastes from reactors or process buildings. The final waste form would be either HLW or remote transuranic waste depending upon the feed material.

Attachment

Attached is a detailed description of the GMODS Process.

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Conversion Of Plutonium Scrap And Residue To Borosilicate Glass Using The GMODS Process

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ABSTRACT

Plutonium scrap and residue represent major national and international concerns because (1) significant environmental, safety, and health (ES&H) problems have been identified with their storage; (2) all plutonium recovered from the black market in Europe has been from this category; (3) storage costs are high; and (4) safeguards are difficult.

It is proposed to address these problems by conversion of plutonium scrap and residue to a CRACHIP (CRitically, Aerosol, and Chemically Incert Plutoium) glass using the Glass Material Oxidation and Dissolution System (GMODS). CRACHIP refers to a set of requirements for plutonium storage forms that minimize ES&H concerns. The concept is several decades old. Conversion of plutonium from complex chemical mixtures and variable geometries into a certified, qualified, homogeneous CRACHIP glass creates a stable chemical form that minimizes ES&H risks, simplifies safeguards and security, provides an easy-to-store form, decreases storage costs, and allows for future disposition options.

GMODS is a new process to directly convert metals, ceramics, and amorphous solids to glass; oxidize organics with the residue converted to glass; and convert chlorides to borosilicate glass and a secondary sodium chloride stream. Laboratory work has demonstrated the conversion of cerium (a plutonium surrogate), uranium (a plutonium surrogate), Zircaloy, stainless steel, and other materials to glass. GMODS is an enabling technology that creates new options. Conventional glassmaking processes require conversion of feeds to oxide-like forms before final conversion to glass. Such chemical conversion and separation processes are often complex and expensive.

INTRODUCTION

In the United States, significant ES&H concerns¹ have been identified with the storage of plutonium scrap and residue. A similar situation is thought to exist in Russia. All of the plutonium recovered from the black market in Europe has been from this category. Storage costs are high and safeguards are difficult. These difficulties are a direct result of the characteristics of these materials. Plutonium scrap and residue normally consist of only a few weight percent plutonium, but the total volume and mass exceed that of clean plutonium. The materials have highly variable chemical and nuclear characteristics. Some of the chemical forms are hazardous and corrode their containers.

It is proposed to address these problems by conversion of plutonium scrap and residue to a CRACHIP (CRitically, Aerosol, and Chemically Incert Plutoium) glass using the Glass Material Oxidation and Dissolution System (GMODS). CRACHIP refers to a set of requirements² for plutonium storage forms that minimize ES&H storage and transport risks. Conversion of plutonium from complex chemical mixtures and variable geometries into a certified, qualified, homogeneous CRACHIP glass with fixed

dimensions in standard containers (1) creates a stable chemical form that minimizes ES&H risks, (2) simplifies safeguards and security (number count safeguards), (3) provides an easy-to-store form, (4) decreases storage costs, and (5) allows for future disposition options.³

GMODS is a new process^{4,5} for directly converting scrap and residue into glass. Earlier glassmaking processes required that plutonium feed material first be a relatively pure oxide-like material before being converted to glass. This requirement implied a complex processing step to yield an oxide form acceptable for conventional glass melters. The technical and economic difficulties in conversion of plutonium scrap and residue to CRACHIP glass have been major barriers for this treatment option. The objective of GMODS development is to provide a low cost, technically feasible process to make CRACHIP glass.

A NEW APPROACH TO PLUTONIUM STORAGE AND DISPOSITION

Before any option to treat plutonium scrap and residue can be undertaken, the requirements for the anticipated product must be defined. A CRACHIP glass must (1) be mechanically stable and must not form aerosols under storage or accident conditions, (2) be chemically inert, and (3) contain sufficient neutron poisons to prevent nuclear criticality with any quantity of material and/or any geometry. This addresses the near-term ES&H issues. This glass must also allow multiple disposition options: long-term storage, recovery of plutonium (with some difficulty), and disposal of plutonium as a waste. In the intermediate term, a CRACHIP form minimizes storage costs and ES&H storage risks, and simplifies safeguards.

The requirements and criteria for CRACHIP glass are similar to those required for high-level-waste (HLW) glasses. Radioactive wastes become less hazardous with time; hence, the fundamental concept in waste management is to isolate (store) these wastes until they are nonhazardous. Glass has been chosen worldwide as the preferred HLW storage and transport form because of several of its properties: (1) acceptance of impure feeds, (2) low solubility in water, (3) chemical inertness, (4) acceptable mechanical integrity, (5) ability to handle high heat loads from decay heat, and (6) avoidance of nuclear criticality by use of neutron poisons. The similar requirements of waste management and plutonium scrap and residue management provide the basis for defining performance requirements for CRACHIP glass: storage with performance equivalent to that of HLW glass.

Several groups are developing optimum compositions^{6,7} for high-plutonium-loaded glass. For plutonium scrap and residue, traditional HLW glass compositions may also be modified for the plutonium and other components in the feed. In this case, plutonium is a minor component in the glass. Glass compositions must be optimized to accept both the plutonium and the other components in the feed.

Regardless of the long-term disposition of plutonium scrap and residue, storage is the only viable near-term option. This implies that the near-term incentive for conversion of plutonium scrap and residue to any storage form is to minimize storage costs. CRACHIP glass reduces the *storage requirements* and, in turn storage costs for plutonium scrap and residue by the following mechanisms:

- **Nuclear Criticality.** Plutonium is currently stored in vaults in small containers (traditionally <5 kg of plutonium per container) that are widely spaced to avoid nuclear criticality. CRACHIP glass with

neutron poisons eliminates criticality control as a vault requirement and thus reduces the vault size. In large vaults, most of the space is empty for geometric criticality control and can be eliminated if the material is stored as CRACHIP glass.

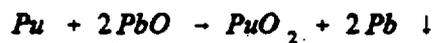
- *Volume Reduction.* Conversion of heterogeneous materials to a high-density, monolithic glass further minimizes storage costs.
- *Safeguards.* Conversion of highly heterogeneous materials to homogeneous glass in a standardized package allows for (1) more precise and reliable safeguards, (2) automated systems, and (3) number-count safeguards. This minimizes the costs of safeguards.

THE PROCESS: GMODS

Conversion Of Metals, Ceramics, Amorphous Solids, And Organics To Glass

GMODS converts plutonium and the other elements within the scrap and residue directly to borosilicate glass. GMODS is a batch process (Fig. 1) during which sequential process steps convert feeds to glass. The initial condition for the process is a melter filled with a molten oxidation-dissolution (lead borate) glass, which has a composition of 2 or more moles of lead oxide (PbO) per mole of boron oxide (B₂O₃). The PbO is a component of the glass and a sacrificial oxide. The process consists of the following steps:

- *Addition of feed material to the molten dissolution glass (Fig. 1.b).* The ceramic (plutonium oxide (PuO₂), etc.) and amorphous components in the feed dissolve into the glass. While metals and organics do not dissolve into conventional molten glasses, the GMODS dissolution glass has special properties to process these materials *in situ*. The inclusion of the sacrificial oxide—PbO—in the molten glass provides a method to oxidize *in situ* (a) metals to metal oxides and (b) organics to carbon dioxide (CO₂) gas and steam. When plutonium or another metal is fed to the melter, it is converted to a metal oxide. These metal oxides dissolve into the glass; carbon oxides (in gaseous form) and steam exit the melter. The reaction product, molten lead, separates from the glass and sinks to the bottom of the melter to form a separate layer,



- *Addition of glass additives [silicon oxide (SiO₂) etc.] to improve the product quality (Fig. 1.c).* The optimum compositions of glasses for rapid oxidation-dissolution of materials in molten glass are different in composition from those for long-term durability; thus, additives that create a more durable glass are introduced after feed oxidation-dissolution takes place.

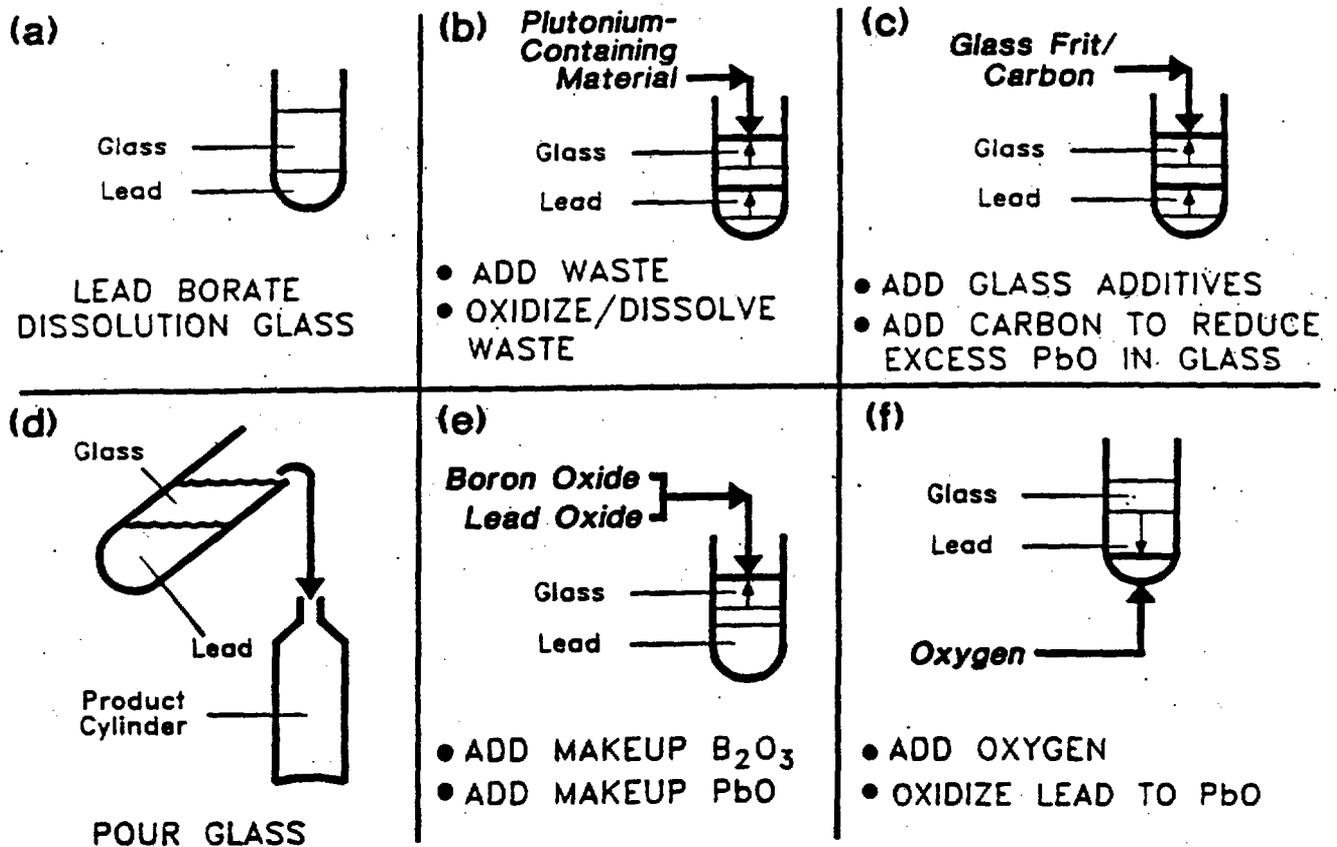
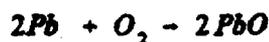


Fig. 1. GMODS batch processing of plutonium-containing material to borosilicate glass.

- *Addition of carbon to remove excess PbO (Fig. 1.c).* Carbon reduces the PbO to lead metal while producing gaseous CO_2 . Excess PbO is removed from the dissolution glass for multiple reasons: (1) more durable glass, (2) reduction of the volume of glass, and (3) avoidance of the costs to provide added sacrificial PbO. The final glass may contain some or no lead, depending on the final desired glass composition.
- *Pouring glass from the furnace followed by solidification (Fig. 1.d).*
- *Addition of B_2O_3 and PbO, as needed, to the melter for processing the next batch of materials (Fig. 1.e).*
- *Reoxidation of the lead at the bottom of the melter to PbO by addition of oxygen (Fig. 1.f).* This oxidation creates the new dissolution glass for the next batch of feed to be processed. Lead is an oxygen carrier that does not leave the system. The oxidation reaction is



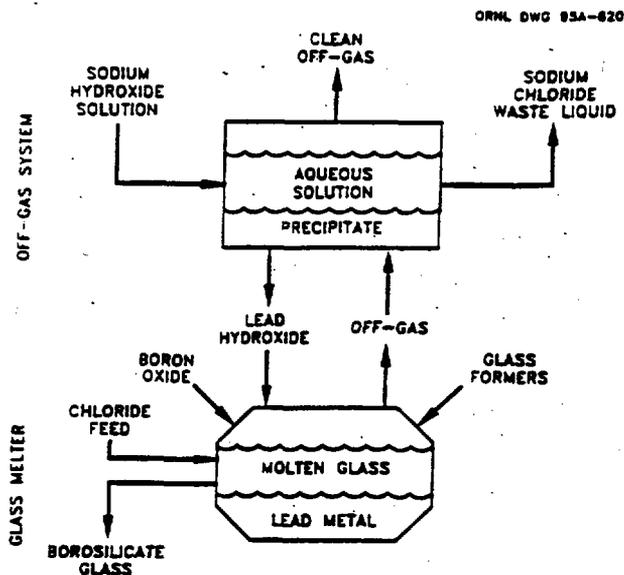


Fig. 2. GMODS processing of chloride-containing feed materials.

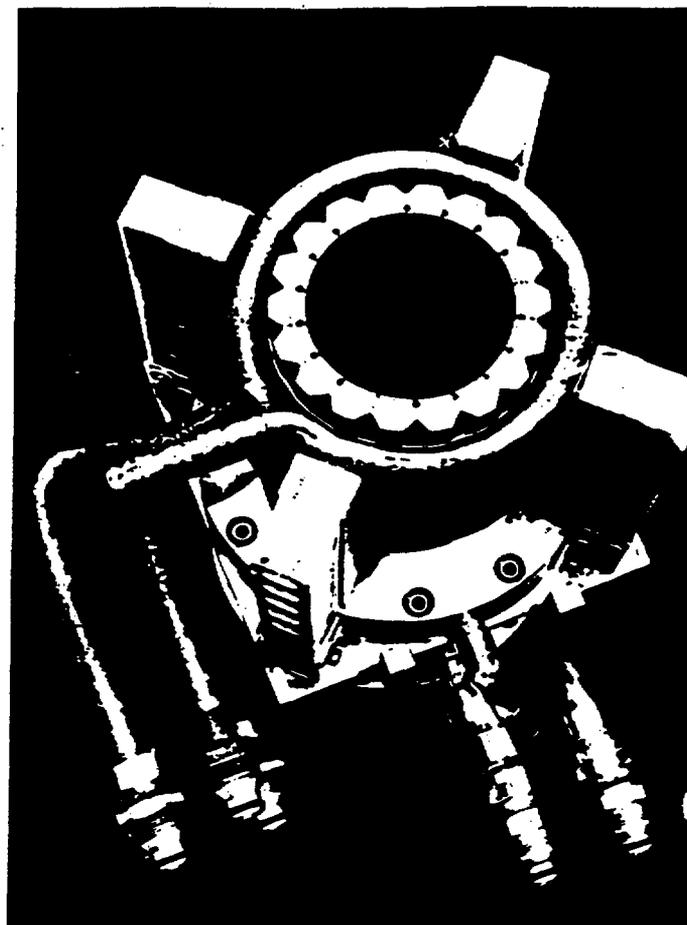


Fig. 3. Small commercial, cold-wall induction-heated melter (Courtesy of Consarc)

Conversion Of Chlorides to Low-Chloride Borosilicate Glass And A Secondary Sodium Chloride Stream

GMODS is designed to convert chloride-containing plutonium residues to glass and create a separate nonradioactive sodium chloride (NaCl) waste stream. Halogens, such as chloride, make poor-quality storage forms; hence, they must be separated from other components in plutonium residues. The analogy used in waste management is that good storage forms (silica, titanates, etc.) for radioactive materials can be found at any ocean beach. Materials that dissolve in seawater (chlorides, etc.) make poor storage forms.

The separation process for chlorides is shown in Fig. 2. In the dissolution glass, chlorides in the feed form lead chloride ($PbCl_2$), which is volatile at glass melter temperatures and exits to the aqueous sodium hydroxide (NaOH) scrubber. In the scrubber, the $PbCl_2$ reacts with the NaOH to yield insoluble lead hydroxide [$Pb(OH)_2$] and soluble NaCl salt. The insoluble $Pb(OH)_2$ is recycled back to the melter where it decomposes to PbO and steam, while the aqueous salt stream (NaCl) is cleaned and discharged as a chemical waste.

Conversion Of Heterogeneous, Poorly Characterized Feed Materials To High-Quality Homogeneous Glass

A certified, qualified, high-quality, homogeneous glass product is required. Thus, an approach similar to that used to produce speciality metals and glass is used, where scrap and residue are fed to the melter and a homogeneous molten glass solution is produced. With a homogeneous glass solution, composition can be determined by limited sampling using mass spectrometric analysis. From the chemical analysis, the required compositions of additives can be determined to produce an appropriate product glass. This strategy, which depends on the ability to create a homogeneous glass from poorly characterized feed materials, is achieved by two mechanisms:

- GMODS can accept wide variations in the chemical composition of the feed and convert the materials to a homogeneous glass. This capability is a prerequisite because it avoids the need for detailed sampling of feed materials to ensure processability.
- The GMODS melter uses process tomography instrumentation^{8,9} to determine in real-time when a homogeneous glass solution has been created. Homogeneous solutions imply homogeneous radiation fields that instrumentation can detect. With variable feeds, dissolution times will vary widely. Instrumentation ensures homogeneous feeds without requiring that tests be conducted on every feed to determine required dissolution times.

EQUIPMENT

The primary GMODS equipment is an induction-heated, cold-wall melter (Fig. 3 shows a small commercial type), which is required because of the corrosive characteristics of the initial dissolution glass. Cold-wall melters have cooling jackets in the wall to produce a "skull" of solidified material that protects the wall from the melter contents. They are used to melt high-temperature materials (e.g., titanium and superalloys) and to produce ultrapure materials (e.g., glass for fiber optics). Russia, France, and the United States are modifying such equipment for processing various radioactive wastes. Batch size may be as large as several hundred kilograms for plutonium scrap and residue with low plutonium concentrations. In Europe, cold-wall melters are currently being developed for throughputs of up to 800 kg/h - far in excess of the size required for this mission.

STATUS OF DEVELOPMENT

Investigations of Process Steps

Some steps of the GMODS process are new, while others are parts of standard industrial processes. Experiments were performed to understand and prove the unique features of GMODS. Literature searches have been conducted to understand those parts of the process that are used in other industrial processes. Each step has also been accomplished in our laboratory.

Laboratory experiments were conducted in platinum and high-fired aluminum oxide crucibles within vertical tube furnaces. Platinum was used for experiments that did not involve lead (lead dissolves into platinum at high temperatures). Various ceramic crucible materials were investigated for use in oxidation process experiments. While the dissolution glass dissolves oxides, the rate of dissolution with CoorsTM high-fired aluminum oxide crucible is sufficiently low for short-time experiments.

A typical experiment involved several hundred grams of material, with uranium and cerium being used as plutonium surrogates. Plutonium tests have been proposed. The plutonium content of scrap and residue is, at most, a few weight percent; hence, in terms of chemical processing, plutonium is a minor component.

Addition of feed material to the molten dissolution glass (Fig. 1.b). The addition of feed materials involves oxidation, dissolution, and mixing of feeds with the molten dissolution glass. Each of these steps has been investigated.

Tests demonstrated the dissolution of UO_2 , ZrO_2 , Al_2O_3 , Ce_2O_3 , MgO , and other oxides. The glasses were examined by a variety of methods to ensure complete dissolution. As expected, the high-boron oxide glass melt had good dissolution capabilities for oxides. In analytical chemistry, B_2O_3 is the standard chemical reagent for fusion dissolution of unknown oxides because of its capability to dissolve such materials. Boron oxide is also the key component in many welding fluxes, which are used to dissolve iron oxides into a glassy slag during the welding process so that they are not incorporated into the weld.

Oxidation-dissolution tests demonstrated the oxidation of the following metals and alloys followed by the dissolution of their oxides into the melt: U, Ce, Zircaloy-2, Al, stainless steel, and other metals. Figure 4 shows cerium glass and lead by-product from a test of oxidation of cerium metal (plutonium surrogate).

Oxidation-dissolution tests also demonstrated the oxidation of carbon and graphite, with production of CO_2 . For centuries, lead oxide has been used to oxidize organics¹⁰. It is the basis for the fire assay method for recovering noble metals (primarily gold) from silicate rock. Lead oxide, various organics, and silicate rocks are mixed together and heated. As the mixture melts, the lead oxide is reduced to metal by the organic. The noble metals in the molten mass then dissolve into the lead, which forms a separate layer that sinks to the bottom. This layer is then processed to separate the noble metal from the lead.

Limited chloride dissolution tests with NaCl demonstrated that lead exits the dissolution glass as PbCl_2 , thus providing a separation of the chloride from other materials. This is a major mechanism for lead to escape from processes where lead and chlorides coexist at high temperatures.¹¹ The basic chemistry is well understood.

Experimental measurements were made of the viscosity of the dissolution glass with various added materials. Experience in the glass industry indicates that molten glass viscosities should be below 100 centipoise (about the viscosity of olive oil) for good mixing and creation of homogeneous glasses. Based on our experimental data, the GMODS dissolution glass temperature will need to be between 800 and 1000°C. The final processing temperature after addition of the silica will be above 1000°C because this addition increases glass viscosity.

Addition of glass additives [silicon oxide (SiO_2) etc.] to improve the product quality (Fig. 1.c). This process step is essentially identical to that used for producing many specialty glasses.¹²

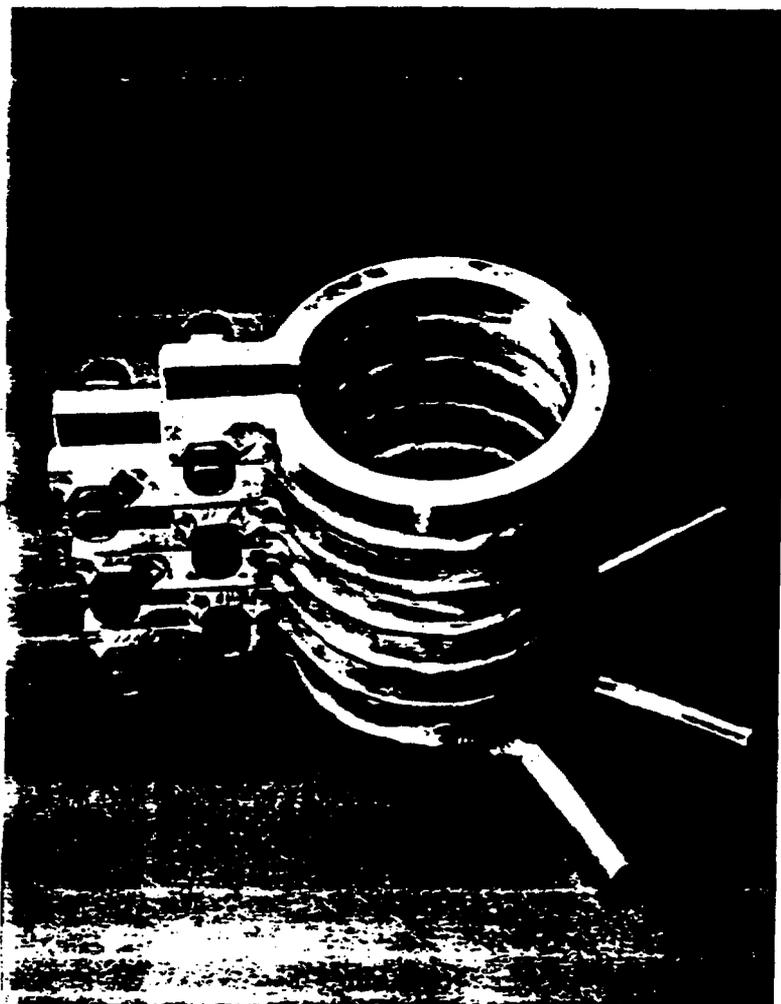
Cerium Loaded Glass



Fig. 4. Cerium glass and lead metal from completed oxidation-dissolution test with cerium metal.

Lead Reaction Product

Fig. 5. ORNL single-turn, cold-wall, induction-heated glass melter.



Addition of carbon to remove excess PbO (Fig. 1.c). This process step is used in several lead-smelting processes, such as the QSL process, to recover lead metal from lead oxide in molten slag.¹³ This step has also been demonstrated with HLW glass in hot cells for recovery of fission product noble metals at Pacific Northwest Laboratory,¹⁴ using a modification of the fire-assay method described above. Because some proposed plutonium glasses are variants of HLW glasses, this experience is particularly relevant.

Pouring glass from the furnace followed by solidification (Fig. 1.d). This is a standard operation used by the glass industry.

Addition of B₂O₃ and PbO, as needed, to the melter for processing the next batch of materials (Fig. 1.e). This is a standard operation used by the glass industry for producing lead borosilicate glass (fine crystal).

Reoxidation of the lead at the bottom of the melter to PbO by addition of oxygen (Fig. 1.f). This is one of several processes used for producing lead oxide for batteries and other uses.¹⁵

Flowsheet Analysis

An analysis of GMODS was performed using the process simulator FLOW.¹⁶ The simulator includes a set of rules to choose glass compositions that meet process (viscosity, etc.) and performance requirements, using Savannah River HLW glass as a basis. The analysis identified critical process parameters when processing plutonium scrap and residue to a borosilicate glass, which is designed to be equivalent in performance to HLW glass. The two key observations were as follows:

- Incentives exist to maximize the GMODS operating temperature. For example, processing 1 kg of plutonium-containing chloride salt residues at 1103° C yields 6.5 kg of glass under standard conditions. Allowing the processing temperature to increase to 1167° C reduces the final glass quantities to about 3 kg. In this case, the waste loading in the glass is limited by the need to minimize molten glass viscosity during process operations to ensure good glass mixing. Increasing the processing temperature decreases the glass viscosity, minimizes the need to add sodium oxide to lower glass viscosity by changing the chemical composition, and allows a higher waste loading in the product glass. With the use of cold-wall, induction-heated melters that are not temperature-limited, the limitation on the maximum process temperature is volatilization of selected glass components.
- Incentives also exist to blend different feeds to minimize glass volumes. For example, blending plutonium salt and ash residue streams and converting them to glass reduces the final volume of glass by about 50% as compared with separate conversion of the two materials to glass. Final glass volumes are minimized because the ash stream provides necessary silica and aluminum to the final product glass, while the chloride stream provides necessary sodium to the final product glass.

The flowsheet simulator also afforded a bounding estimate of glass quantities if scrap and residue at Rocky Flats were converted to glass by GMODS. For "lean" scrap, 232 m³ of glass would be produced with an average plutonium content of 0.06%. For "rich" scrap, 34 m³ of glass would be produced with

an average plutonium content of 2.4%. The actual quantities of glass may be significantly less when higher process temperatures are used.

The quantities of glass produced from processing plutonium scrap and residue are not determined by the plutonium (due to its low concentration) but, rather, by other components in the scrap or residue. The set of constraints includes: glass processing temperatures, solubility limits of specific elements in the product glass, and glass durability under repository conditions. If the goal is to minimize glass volumes in order to minimize storage or disposal volumes, selected pre-processing of some residues can be undertaken to minimize specific elements that most impact glass volumes. This choice involves a series of trade-offs between multiple processes. (Note that organics and chlorides in feeds have little impact on final glass volumes because GMODS is a separations, as well as a glassmaking process.)

Equipment

In making a survey of cold-wall, induction-heated melters, the range of operating conditions was found to substantially exceed the potential range of GMODS operations. Industrial melters are designed to operate at up to 2500°C; some of them operate with molten metal and slag. Recently, a small experimental melter has been built in our laboratory (Fig. 5) to provide a better understanding of this technology. Tests of this melter are under way.

Development Perspective

The analytical testing and laboratory development work have demonstrated each step required for GMODS and identified equipment, instrumentation, and other components required for GMODS. A significant effort, however, will be required to convert GMODS into an industrial technology. This effort will include a better understanding of the process, integration of process steps into a system, and development of equipment.

SUMMARY

GMODS is a new process for the direct conversion of plutonium scrap and residue to CRACHIP glass. It is designed to (1) convert metals, ceramics, and amorphous solids to glass; (2) oxidize organics with conversion of residues to glass; and (3) convert chlorides into a chloride-free borosilicate glass and a secondary clean NaCl stream. GMODS is an enabling technology, since it creates new plutonium scrap and residue management options. Because these options address common national security, non proliferation, and ES&H concerns, they may be acceptable to both the United States and Russia. As a new technology, however, GMODS has significant technical uncertainties that must be resolved in additional studies.

DISCLAIMER

The views expressed in this paper are those of the authors and do not necessarily reflect any biases, proposed actions, or decisions of the United States Government or any agency thereof.

REFERENCES

1. *Plutonium Working Group Report on Environmental, Safety, and Health Vulnerabilities Associated with the Department's Plutonium Storage*, DOE/EH-0415, U.S. Department of Energy, Washington D.C., November 1994.

2. Forsberg, C. W. *What Is Plutonium Stabilization and What Is Safe Storage of Plutonium?*, ORNL/M-4322, Lockheed Martin Energy Systems, Inc., Oak Ridge National Laboratory, Oak Ridge, Tennessee., June 1995.
3. *Management and Disposition of Excess Weapons Plutonium*, National Academy of Science, National Academy Press, Washington D.C., 1994.
4. Forsberg, C. W.; Beahm, E. C.; Parker, G. W., *Radioactive Waste Material Disposal*, U.S. Patent 5,461,185, Office of Trademarks and Patents, U.S. Department of Commerce, Washington D.C., October 24, 1995.
5. Forsberg, C. W.; Beahm, E. C.; Parker, G. W.; Rudolph, J.; Haas, P.; Malling, G. F.; Elam, K. R.; Ott, L., *Direct Vitrification of Plutonium-Containing Materials With The Glass Material Oxidation and Dissolution System (GMODS)*, ORNL-6825, Lockheed Martin Energy Systems, Inc., Oak Ridge National Laboratory, Oak Ridge, Tennessee, October 1995.
6. Ramsey, W. G.; Bibler, N. E.; and Meaker, T. F.; "Compositions and Durabilities of Glasses for Immobilization of Plutonium and Uranium," *Proceedings of Waste Management '95*, Tucson, Arizona, March 1, 1995, American Nuclear Society, La Grange, Illinois, 1995.
7. Bates, J. K.; Ellison, A. J. G.; Emery, J. W.; Hoh, J. C. "Glass as a Waste Form for Immobilization of Plutonium," *Proceedings of the Materials Research Society, Boston, Massachusetts*, November 27, 1995.
8. Shanley, A., "Process Tomography: Seeing Is Believing," *Chem. Eng.* October 1995, 102(10):30.
9. Idaho National Engineering Laboratory, *Proceedings of the 4th Nondestructive Assay and Nondestructive Examination Waste Characterization Conference*, Salt Lake City, Utah, CONF-951091, Idaho Falls, Idaho, October 24-26, 1995.
10. Ercker, L., *Treatise on Ores and Assaying*, ~1500.
11. Linak, W. P. and Wendt, J. L., "Toxic Metal Emissions From Incineration: Mechanisms and Control," *Prog. Energ. and Combust. Sci.*, 19:145, 1993.
12. McKinnis, C. L, and Sutton, J. W., "The Glassmaking Process I: A Theory of the Nature of Silicate Melts and Their Interaction with Silica," *J. Am. Ceram. Soc.* 42(4):194, April 1959.
13. King, M., "Lead," in *Encyclopedia of Chemical Technology*, eds. J. I. Kroschwitz and M. Howe-Grant, John Wiley & Co., New York, 1995.
14. Jensen, G. A.; Platt, A. M.; Mellinger, G. B.; Bjorklund, W. J., "Recovery of Noble Metals From Fission Products," *Nucl. Technology*, 65:305, May 1984.

15. Carr, D. S.; Spangenberg, W. C.; Chronley, K., "Lead Compounds," in *Encyclopedia of Chemical Technology*, eds. J. I. Kroschwitz and M. Howe-Grant, John Wiley & Co., New York, 1995.
16. Ferrada, J. J.; Nehls, J. W.; Welch, T. D.; Giardina, J. L.; Maliyekkel, A. T.; Forsberg, C. W., *Preliminary Process Simulation and Analysis of GMODS: Processing of Plutonium Scrap and Residue to Glass*, ORNL/TM-13109, Lockheed Martin Energy Systems, Inc., Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1995.

Dissolve and Vitrify

Introduction

In the Dissolve and Vitrify process, the SNF is dissolved electrolytically and depleted uranium is added to reduce the enrichment of the waste form. Glass forming chemicals are added to the resulting solution, and then vitrified directly.

Process

The process is as follows:

- a) The SNF is added to an electrolytic dissolver of the type currently contained in the SRS H-Canyon. The solution in the dissolver is nitric acid saturated with boric acid. This system is run as a batch process. The uranium concentration in the solution is maintained below 10 g/L for nuclear safety.
- b) The resulting solution containing dissolved SNF will be transferred to a holding tank. Chemical and isotopic analyses will be performed. The solution will be adjusted (e.g., to a desired solids concentration) and then transferred to a melter feed tank. Unless it is demonstrated to be mandatory for proper mixing, dissolver feed solution and glass forming chemicals (or frit) will not be mixed until the vitrification step.
- c) The feed solution and glass forming chemicals will be fed to a joule-heated melter and fused into a glass. The melter can be sized to process the feed from at least four electrolytic dissolvers of the type at SRS. The molten glass will be poured out of the melter into stainless steel canisters. The glass product will be designed so that it meets all requirements for acceptance at a geologic repository.

Advantages

All aspects of the treatment have been demonstrated at full production scale at the Savannah River Site. Uncertainties are minimized by reliance on proven, mature technology. Secondary wastes can be treated as high level waste. The similarity of the waste form to that qualified for high level waste is expected to reduce the qualification burden.

Attachment

Attached is an overview of the Dissolve and Vitrify treatment process.

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**Vitrification of Spent Nuclear Fuel
Executive Overview**

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December 8, 1995

Overview

This document describes technologies which could be used to vitrify spent nuclear fuel. Two viable processes are described. Fuel could be electrolytically dissolved in acid solution and then mixed with glass forming chemicals and vitrified in electric furnaces or melters. An alternate approach would be to dissolve the fuel in molten lead borate and then add glass forming chemicals to make the final glass, which would be done in an induction melter. The processes and their relative advantages and disadvantages are described. It is recommended that both options be pursued to the extent that facility, production, and storage costs for each process may be accurately estimated.

Summary

Two technologies for dissolving and then vitrifying the spent nuclear fuel stored at the Receiving Basin for Off-site Fuel (RBOF) at the Savannah River Site are described. The processes would each convert spent fuel to a glass which shall meet all acceptance requirements for high-level waste. No apparent reasons were identified which preclude either option from being performed in existing "Hot" canyon facilities.

One process option would be to utilize existing fuel dissolution technology. There are several existing processes for dissolving spent fuel. One potential method (used as a basis for this report) would be to charge spent fuel to an electrolytic dissolver (same type as located in the SRS H-canyon) converting it to an acid solution, predominately containing aluminum, uranium, and boron. This solution would then be mixed with glass forming chemicals and vitrified into high-level waste glass by a commercial joule heated melter. Both ends of this process are based on very mature and well known technology.

Another, potentially more direct method for dissolving and vitrifying spent fuel is the Glass Material Oxidation and Dissolution System (GMODS) process, a new technology developed by the Oak Ridge National Laboratory. Spent fuel would be fed into a molten solution of lead oxide and boric oxide. The lead oxide would oxidize the metallic components (mostly cladding). The resulting alumina and uranium oxide would then be dissolved into the borate (the lead being greatly depleted) system. The molten solution would then be mixed with glass forming chemicals to make high-level waste glass. As a new technology, GMODS is a less certain process, but does offer the added potential for processing a very large range of miscellaneous high-activity wastes in addition to spent fuel.

While the two options have obvious differences, both dissolution steps result in a solution which is dominated by boron, aluminum and uranium. This will probably allow the glass formulations (and the resulting waste acceptance process) to be quite similar, independent of the dissolution process. Further, this should allow the two processes to be more closely and evenly evaluated as opposed to completely divergent technologies.

Introduction

The Savannah River Site (SRS) will likely be required to manage up to 34,000 Material Test Reactor equivalents (MTRE) of spent nuclear fuel. Depending on the outcome of current National Environmental Policy Act (NEPA) activities, the storage or processing of spent fuel at SRS may continue through the year 2035. The current management plan for the SRS fuel (predominately Al clad fuel) investigates several processing and storage options which serve as input to the NEPA activities. This document was written at the

request of the DOE to provide additional options for dissolving fuel and immediately processing into high-level waste glass. The primary goal was to identify feasible processes which could be performed in either existing facilities or in additions adjunct to existing facilities.

The following sections provide a brief description of each process. The expected glass production rate and waste loading are estimated. Minimum development activities are identified and then conclusions are discussed.

Process Descriptions

This section provides a brief description of the primary chemical processing steps which will be necessary for either vitrification option. The product glass form and the expected production rate are discussed. Detailed estimates of process costs for either option can be generated on an equivalent basis as soon as the GMODS process is developed to the degree described in the Path Forward section.

Option 1. Traditional Fuel Dissolution Followed by Vitrification

Spent nuclear fuel could be dissolved and vitrified at either F or H canyon at the SRS. The process described here is based on electrolytic dissolution followed by vitrification in a commercial joule heated melter. Electrolytic dissolution was selected as a basis for discussion since it avoids the addition of mercury (which is a part of Mercury-Nitric Acid dissolution method) or the need to dilute the enriched uranium with depleted uranium (as required by a 12 step dissolution method used in F-Area). Mercury is not compatible with vitrification processes and would need to be removed prior to the vitrification step. Dilution of the uranium would greatly increase the total amount of glass to be made. The electrolytic dissolver (of the type at the SRS H-canyon) was designed to handle enriched fuel and does not require mercury as a dissolution aid. It is also capable of processing Zr clad fuel. The process for converting the spent fuel to glass would consist of the four primary chemical processing steps described below.

- 1) Spent nuclear fuel is added to the electrolytic dissolver (each dissolver could process on the order of 27kg aluminum per day). The solution in the dissolver is nitric acid saturated with boric acid. This system is run as a batch process and the U concentration in solution will be maintained below 10g/L. *Electrolytic dissolution and the processing cycles are detailed in "Processing of Irradiated Enriched Uranium Fuels at the Savannah River Plant," and the associated references.*

The fuel will contain an aluminum to uranium mass ratio between 6 to 60 and the nitric acid solution also contains considerable boric acid. The total dissolved solids of the solution will be of sufficient concentration that minimal solution adjustment is expected prior to delivery to the melter.

- 2) The resulting dissolver solution will be transferred to a holding tank and a sample will be pulled for chemical and isotopic analysis. The solution will be adjusted (if necessary) and then transferred to a melter feed tank. Unless it is demonstrated to be mandatory for proper mixing, dissolver feed solution and glass forming chemicals (or frit) will not be mixed until the vitrification step.

- 3) The feed solution and the glass forming chemicals (or frit) will be delivered to the melter from separate sources and mixed together in a single feed line just prior to being fed onto the melt cold cap. The melter will be a joule heated unit, probably a close derivative of the commercial melter undergoing testing as part of the Transportable Vitrification System (TVS). This melter can be sized to process the feed from 4 or more electrolytic dissolvers - i. e. the TVS unit is capable of producing up to 65kg / hour glass from a wet sludge feed. Another possible option is an induction melter (the type required for GMODS).
- 4) The feed solution and glass forming chemicals are fused into a glass which meets all high-level waste acceptance requirements and the glass is then poured from the furnace and allowed to solidify.

Option 2. Glass Material Oxidation and Dissolution System (GMODS)

GMODS is a process by which spent nuclear fuel is directly converted to a high-level waste glass product. GMODS can directly:

- 1) convert metals, oxides, and amorphous solids to borosilicate glass,
- 2) convert chlorides to glass with production of a secondary non-radioactive sodium chloride chemical waste stream,
- 3) oxidize organics with residues converted to glass.

Conventional vitrification processes have limited ability to handle metals, chlorides and organic bearing materials. GMODS, therefore, can directly process a broader range of materials than conventional vitrification processes. Spent nuclear fuel typically consists of metal or oxide fuel with metal clad and requires chemical pretreatment prior to vitrification (as described above). GMODS includes a chemical pretreatment step and initial testing at SRS indicates the GMODS product can indeed be mixed with glass forming chemicals to form a glass which will meet the acceptance criteria for high-level waste. The GMODS process is based on converting spent nuclear fuel to glass through three sequential batch operations. The initial condition for the process is a melter filled with molten lead borate. The starting molar ratio of PbO to B₂O₃ is 2 or more (≥2). This is a very fluid and corrosive melt. The process consists of the steps described below.

- 1) Spent nuclear fuel is added to the molten lead borate. The ceramic (uranium oxide, etc.) and amorphous components quickly dissolve into the melt. The metallic components are oxidized by the PbO into metallic oxides. These oxides, in turn, dissolve in the melt. This oxidation process (for metallic Al cladding) is written as:



PbO, therefore, is a sacrificial oxide which is converted to metallic lead as a reaction product. Molten lead has a density sufficiently high to sink to the bottom of the reaction vessel. The remaining lead borate and metallic oxide float above the molten lead and the dissolution reactions proceed.

The properties of molten lead borate allow for rapid oxidation and dissolution of metals and ceramic solids. PbO is a powerful oxidant. However, some metals form a protective oxide coating which can slow chemical reactions. B₂O₃ is a solvent of metal oxides and rapidly dissolves such coatings. It is the combination of both PbO and B₂O₃ that makes GMODS feasible.

- 2) After oxidation/dissolution of the spent nuclear fuel, various glass additives (such as SiO₂) are added to the melt. This is done to convert the GMODS melt (which is quite soluble) to a more durable glass form. This glass will be formulated to meet all acceptance criteria for high-level waste.
- 3) The high-level waste glass is poured from the furnace and allowed to solidify.

As necessary, all or some of the PbO may be removed from the system during step 2 (by additional reduction reactions). Lead oxide is only left in the glass if (1) it improves performance of the glass or (2) allows higher waste loadings in the glass. The resulting metallic Pb is oxidized and recycled for a new GMODS batch. *The GMODS process and its preliminary facility basis are detailed in, "New Glass Material Oxidation and Dissolution System Facility: Direct Conversion of Surplus Fissile Materials, Spent Nuclear Fuel, and Other Materials to High-Level Waste Glass," and the associated references.*

Glass Product and Production Rates

The glass product form for either process is expected to be a borosilicate glass with a composition tailored to maximize the waste loading. Regardless of glass type, the glass product will meet all acceptance criteria for high-level waste, i. e., be of a quality consistent with Defense Waste Processing Facility (DWPF) glass. It is expected that the concentration of aluminum will be the limiting factor with respect to waste loading in the glass. The waste loading should be no lower than ≈ 25 percent dissolved fuel by weight (400kg fuel will result in 1600kg glass - or one DWPF canister volume). The primary glass development effort will be to maximize the waste loading (30+ weight percent may be achievable) and glass density as this would greatly reduce the total glass volume created and project cost.

Production of the glass is not expected to be the limiting factor associated with the process. Commercial induction or joule heated melters currently being evaluated for waste vitrification processes have capacities equivalent to (or greater than) the DWPF melter. The limiting factor will be the fuel dissolution. Based on the processing rates discussed above, it appears that the commercial melter with ≈ 65 kg glass per hour output could process the total production of up to four electrolytic dissolvers. GMODS will need to be further evaluated but can likely be scaled to accept batch charges on the order of ten's of kilograms.

Path Forward

An initial GMODS and glass formulation development program are recommended. The program would be a joint effort of ORNL and SRS. The program would be centered around the following activities:

- 1) GMODS Process Development
This activity would include experiments and modeling activities to develop an understanding of processing time (fuel dissolution and glass mixing and melting), auxiliary equipment requirements (off-gas and need for size reduction of feeds), and equipment sizes for a given feed. This data is required input to determine the engineering feasibility and provide an estimate of facility requirements.

- 2) Waste Glass Formulation
This activity would be centered around developing glass formulations (for each process option) which maximizes waste loading and still meets the acceptance requirements for high-level waste. This information is essential for providing the lowest cost process. Glass compositions will also be tested to ensure compatibility with melter materials.

- 3) Facility Analysis
This activity would entail gathering the available information for electrolytic dissolution and the data obtained from Item 1 above (GMODS) and comparing the effective utility and costs for each option (and their potential configurations). The waste form data obtained in Item 2 would then be included to provide basic plant requirements and a cost/benefit analysis. Emphasis would be placed on providing a facility with capability to process the various high-activity waste streams which require remote handling. A cost/benefit analysis comparing existing and new facilities will be provided.

ORNL and SRS would closely coordinate these tasks to ensure the engineering scale development activities and glass formulation work are consistent with the basic requirements and capabilities of nuclear material processing. The majority of effort undertaken in the first six months of the project would be centered around Item's 1 and 2. Two full-time personnel at SRS and two at ORNL would be required. After this initial R&D phase, the facility analysis tasking could begin. The end deliverable would be facility upgrade/cost estimates and initiation of pilot scale operations.

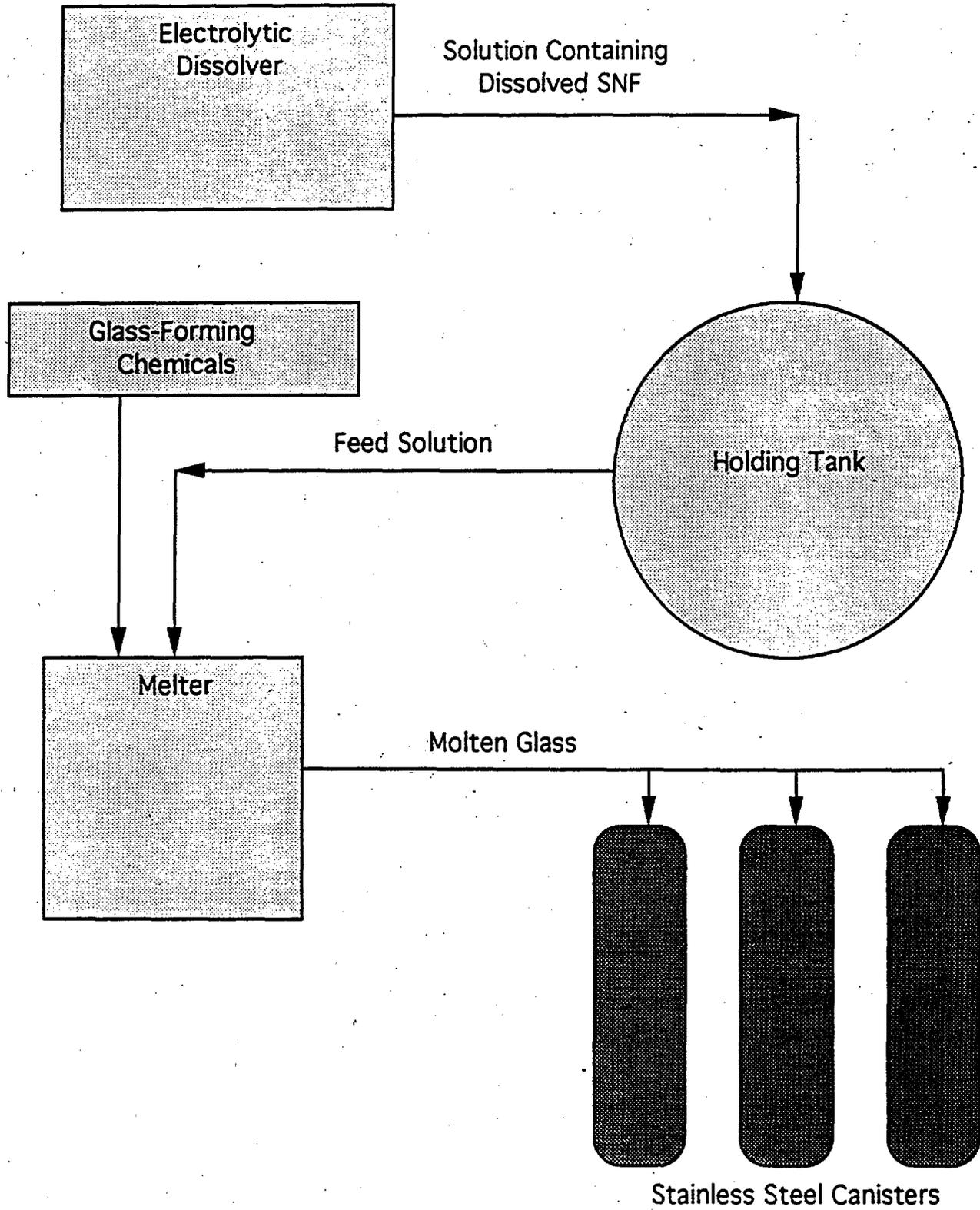
Conclusions

Vitrification of spent nuclear fuel is feasible. Two basic processing options appear to be technically viable and should be further developed to determine the best choice. One process would utilize existing SRS canyon facilities and methods to dissolve spent fuel in acid solution and then process the acid solution into glass using a commercial glass melter system. The other process, GMODS, would directly convert spent nuclear fuel to a high-level waste glass. The more traditional approach involves considerably less uncertainty, but will generate more secondary waste and may require more physical space. The GMODS process (developed at ORNL) has a higher associated process risk due to its early state of development, but offers the greater flexibility towards treating miscellaneous spent fuel and other high activity wastes without major process flowsheet or equipment changes. Both processes will result in a glass form which will meet all acceptance criteria for high-level waste. Oak Ridge National Laboratory and the Westinghouse Savannah River Company are prepared to jointly develop these processes utilizing the complimentary capabilities of their technical personnel and infrastructure.

References

- J. F. Krupa, et al., "Savannah River Site Interim Spent Nuclear Fuel Management Plan - 1995 Update (U)," WSRC-RP-95-670, Westinghouse Savannah River Company, Aiken, SC, 29808, June 1995.
- C. W. Forsberg, et al., "New Glass Material Oxidation and Dissolution System Facility: Direct Conversion of Surplus Fissile Materials, Spent Nuclear Fuel, and Other Materials to High-Level Waste Glass," ORNL/MD/LTR-13, Oak Ridge National Laboratory, Oak Ridge Tennessee, 37831, March 1995.
- M. L. Hyder, et al., "Processing of Irradiated Enriched Uranium Fuels at the Savannah River Plant," DP-1500, E. I. du Pont de Nemours & Co., Aiken, SC, 29808, April 1979.
- C. W. Forsberg, et al., "Direct Conversion of Radioactive and Chemical Waste Containing Metals, Ceramics, Amorphous Solids, and Organics to Glass," proceedings of Spectrum '94 Nuclear and Hazardous Waste Management International Topical Meeting, August 1994.
- J. C. Whitehouse, et al., "Design and Fabrication of a Transportable Vitrification System for Mixed Waste Processing," proceedings of 3rd Biennial Symposium on Mixed Waste, A. A. Moghissi, et al., Ed's, American Society of Mechanical Engineers, pp 8.3.1 - 8.3.8, 1995.
- DWPF Waste Form Compliance Plan, WSRC-IM-91-116-0, Revision 3, Westinghouse Savannah River Company, Aiken, SC, 29808, April 1994.
- W. G. Ramsey, et al., "Relationship Between Borosilicate Glass Composition, Structure and Durability Test Response," Environmental and Waste Management Issues in the Ceramic Industry, Ceramic Transactions, Vol. 39, American Ceramic Society, 1994.
- C.W. Forsberg, et al., "Direct Vitrification of Plutonium Containing Materials (PCMS) with the Glass Material Oxidation and Dissolution System (GMODS)," ORNL-6825, Oak Ridge National Laboratory, Oak Ridge Tennessee, October 1995.

Dissolve And Vitrify



Electrometallurgical Treatment

Introduction

Electrometallurgical Treatment of SNF builds upon an existing technology, and is compact and economical. Facilities and equipment are in place that have sufficient capacity to treat several of the fuel types, and the process is readily adaptable to a wide variety of SNF.

Waste Form

Electrometallurgical Treatment generates a small amount of waste regarding repository disposition, and essentially no secondary low level waste. Its waste form is uniform and predictable, providing for straightforward isotopic dilution of highly enriched uranium.

Attachment

Attached is a document that explains in detail the process and product of this treatment option.

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ELECTROMETALLURGICAL TREATMENT OF ALUMINUM-MATRIX FUELS

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I. INTRODUCTION

The U.S. Department of Energy faces a dilemma concerning spent aluminum-matrix reactor fuel. Over the next forty years, 128 metric tons of spent aluminum-matrix fuel will be shipped to the Savannah River Site from U.S. and foreign research reactors. When originally fabricated, this fuel contained over 55 metric tons of uranium at an average enrichment of ~20%. At Argonne National Laboratory we are developing a cost-effective electrometallurgical process for recovering uranium from these aluminum-matrix fuels. The uranium that can be recovered from this fuel has a commercial value of over \$300 million. Recovering the uranium also reduces the amount of high-level waste that must ultimately be disposed of in a geological repository. Other options for dealing with this fuel are (1) using existing facilities at Savannah River to recover the uranium or (2) disposition of the fuel, including the uranium, in a geological repository. The first option cannot handle all the research reactor spent fuel, because around 2005 the processing canyons at Savannah River will be decommissioned. Direct disposition of the metallic fuel in a geological repository is another possibility, as is disposition following conversion to a glass or ceramic waste form. However it is questionable whether the NRC will allow direct disposal of highly-enriched uranium metallic fuel in the repository. Also, the high level waste volume can be reduced by about a factor of ten by removal of the aluminum and uranium from the fuel by some type of processing. This results in a significant savings in disposal costs.

The electrometallurgical treatment process described in this paper builds on our experience in treating spent fuel from the Experimental Breeder Reactor (EBR-II). The work is also, to some degree, a spin-off from applying electrometallurgical treatment to spent fuel from

the Hanford single pass reactors (SPRs) and fuel and flush salt from the Molten Salt Reactor Experiment (MSRE). In treating EBR-II fuel, we recover the actinides from a uranium-zirconium fuel by electrorefining the uranium out of the chopped fuel. With SPR fuel, uranium is electrorefined out of the aluminum cladding. Both of these processes are conducted in a LiCl-KCl molten-salt electrolyte. In the case of the MSRE, which used a fluoride salt-based fuel, uranium in this salt is recovered through a series of electrochemical reductions. Recovering high-purity uranium from an aluminum-matrix fuel is more challenging than treating SPR or EBR-II fuel because the aluminum-matrix fuel is typically ~90% (volume basis) aluminum.

Thermodynamic calculations predict that electrorefining a uranium-aluminum alloy in a molten chloride electrolyte will not yield a clean separation of uranium from aluminum. To circumvent these difficulties we modified our process. The first modification was changing from a LiCl-KCl electrolyte to a LiF-KF electrolyte. By switching molten salts we are able to electrotransport aluminum to the electrorefiner cathode, leaving uranium and metal fission products behind in the basket of the anode. The aluminum obtained will be high purity and disposed of as low-level waste. Then in a second electrorefiner we will recover high-purity uranium. Fission products will be converted to oxides, then incorporated into a glass that can be fed into the Defense Waste Processing Facility (DWPF) at Savannah River or be incorporated into DWPF-type glass in a small-scale glass melter.

A diagram of the electrometallurgical process is shown in Figure 1. The process will be performed in an inert atmosphere enclosure located in a shielded facility. The process operations are grouped into three types of

operations: (1) head end steps, (2) electrorefining and consolidation steps, and (3) oxidation and glass-forming steps. This paper will discuss the overall process flowsheet with particular emphasis on the

thermodynamics involved in the aluminum and uranium electrorefining steps.

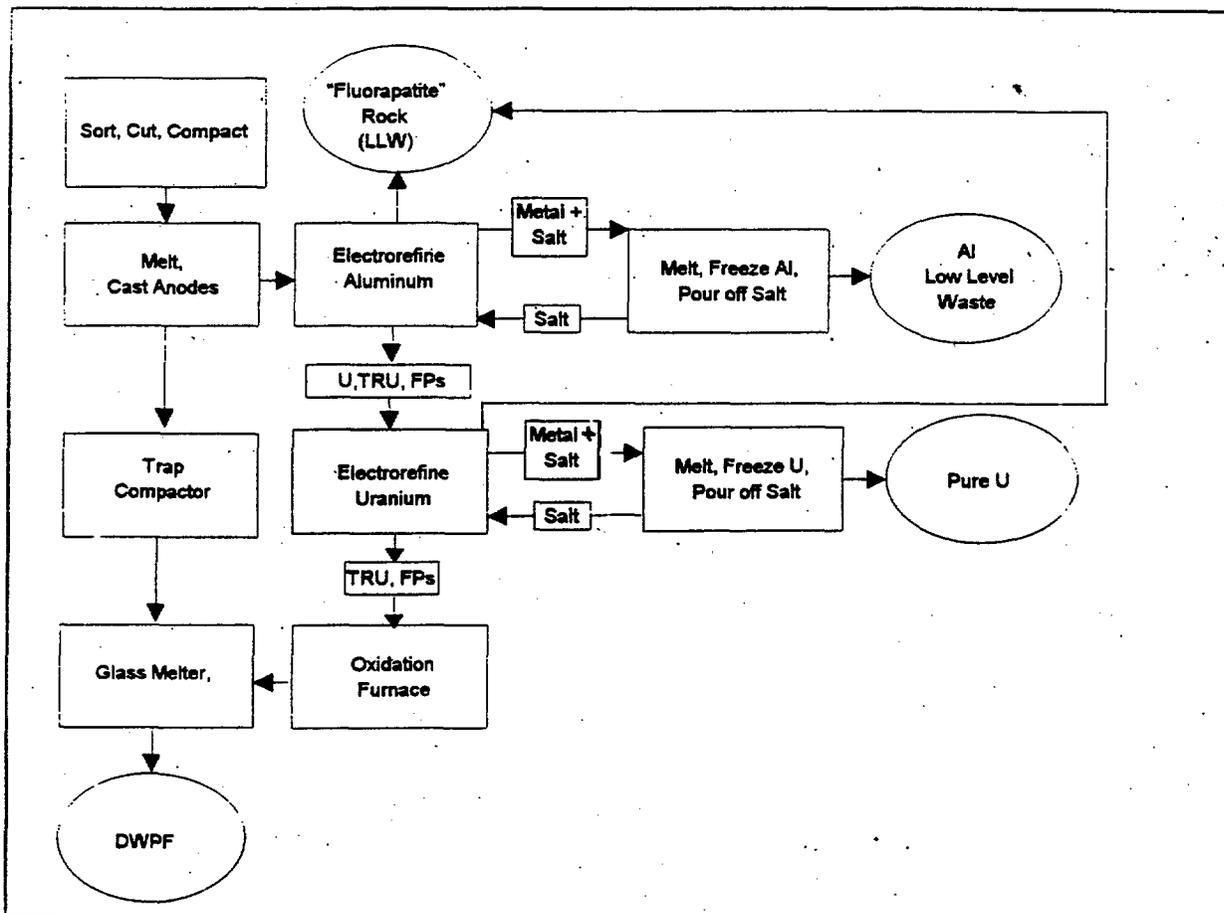


Figure 1. Electrometallurgical Treatment Flowsheet for Aluminum-Matrix Fuels (LLW=low-level waste, TRU=transuranics, Fps=fission products)

II. HEAD-END STEPS

In the first head-end step, the fuel assembly ends are removed, then the fuel is sorted and compacted. Next, the fuel is melted in an enclosed tilt-pour furnace. Because silicon complexes with uranium and enhances the separation of aluminum and uranium, silicon is added to the molten fuel at this point. Then the molten fuel is cast into anodes. In the melting step we expect that the volatile fission products (cesium, rubidium, bromine, iodine, xenon, and krypton) will vaporize. These volatile species, with the exception of xenon and krypton, will be captured in a fibrous aluminosilicate (fiberfrax) trap above the molten metal. We used a similar trapping method to capture volatile metals, iodine, and bromine in the ANL Melt Refining Process and have successfully trapped 100% of volatilized cesium and sodium.¹ After casting of the anodes, the trap material is compressed and added to a glass melter at the end of the process. Because the entire process is conducted in an inert enclosure, xenon and krypton can be vented or trapped in cryogenic traps as part

of the purification system of the inert atmosphere enclosure.

III. ELECTROREFINING AND CONSOLIDATION STEPS

A. General Electrorefining Principles

Electrorefining is used to recover one metal as a pure metal phase from the fuel and leave the more noble metals behind in the anode. Electrorefining in molten halide salts separates metals, based on the relative thermodynamic stability, as metal halides. In general, *the species that form the most stable metal halides are the first to be oxidized at the anode and the last to be reduced at the cathode. The more noble the metal, the less stable the metal halide that is formed.* Metals whose halides are widely separated in terms of thermodynamic stability (>3 kcal/mol F⁻ or Cl⁻) are readily separated. Two conditions must be satisfied to obtain a pure deposit

of a metal (M) at the electrorefiner cathode. First, the voltage of the electrorefiner must be limited to a value such that no metal more noble than M is oxidized at the anode. Second, to obtain a high-purity deposit, the concentration of MX_n (where MX_n is the halide salt of metal M) must be large relative to the other metal halides formed by oxidation of the spent fuel.

In a molten chloride salt, uranium and aluminum cannot be separated from the uranium-aluminum spent fuel. However, by electrorefining, aluminum can be separated out of a uranium-aluminum-silicon alloy by switching to a fluoride salt and by adding silicon to the uranium-aluminum alloy. Figure 2 shows the relative thermodynamic stability of the metal fluorides in a LiF-KF molten electrolyte after the fuel has been alloyed with silicon. Toward the top of the diagram are species that form more stable metal fluorides. As one moves down the diagram, greater voltages (electrorefiner cathode vs. electrorefiner anode) are required to electrotransport the respective metals from the anode to the cathode. The heavy dashed line represents the break between aluminum and rare earth metals. If the electrorefiner is operated at a voltage corresponding to that of the heavy dashed line, only aluminum will be deposited at the cathode.

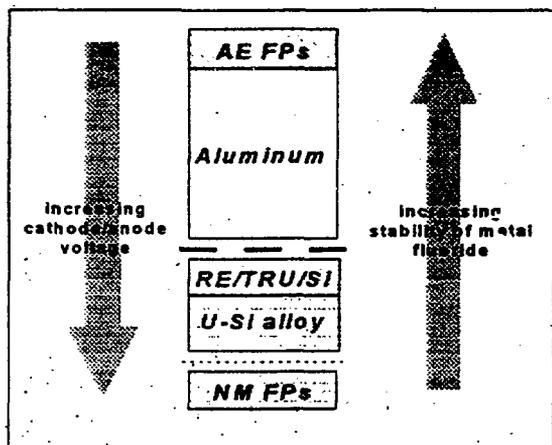


Figure 2. Diagram Describing Electrorefining Separations of Alkaline Earth Fission Products (AEFPs), Aluminum, Rare Earths (RE), Transuranics (TRU), Silicon (Si), and Noble Metal Fission Products (NMFPs)

In aluminum electrorefining, the alkaline earth fission products accumulate in the electrolyte, and the actinides, rare earths, and noble metals remain in the anode. In a similar fashion, in uranium electrorefining the rare earth and transuranic fission products accumulate in the electrolyte, and the noble metal fission products remain in the anode.

Argonne National Laboratory has developed a high-throughput design for both the aluminum and

uranium electrorefiners. A simplified top view of the electrorefiner is shown in Figure 3. The anodes are mounted in a circular array and rotated in the channel between two cylindrical cathodes. Dendritic uranium deposits at the cathode in this design and we anticipate that the aluminum cathode deposit will likewise be dendritic. The dendrites are scraped off the cathode by scrapers attached to the anode baskets. The dendrites then sink to the bottom of the electrorefiner where they are collected. When all of the aluminum or uranium has been electrorefined out of the anodes, the current is turned off, the dendrites are compressed, and the dendrites are removed from the electrorefiner.

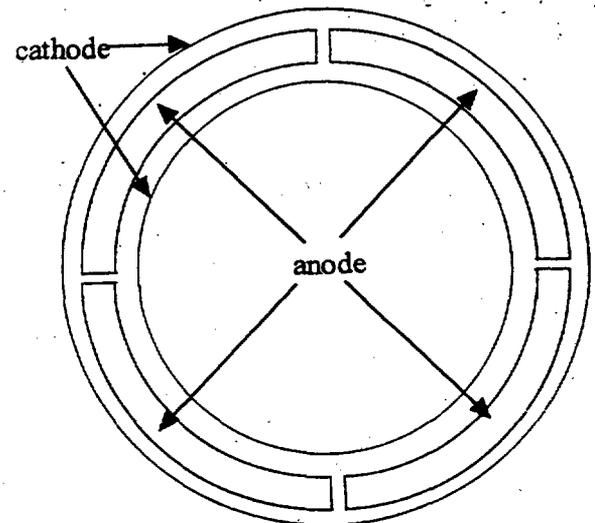


Figure 3. Simplified Top View of High Throughput Aluminum/Uranium Electrorefiner

B. Aluminum Electrorefining

In the aluminum electrorefiner, aluminum metal is oxidized at the anode to form K_3AlF_6 , a species that is soluble in the LiF-KF electrolyte. At the cathode, K_3AlF_6 is reduced to aluminum metal. To obtain a cathode deposit of high-purity aluminum, K_3AlF_6 must be the least thermodynamically stable metal fluoride in the molten electrolyte until all the aluminum has been removed from the anode. To achieve this condition, the cell voltage must be limited to prevent the oxidation of rare earth silicides to rare earth fluorides, yet it must be sufficient to electrorefine all the aluminum. When the activity of aluminum in the anode has been decreased to 10^{-4} , the theoretical equilibrium cell voltage is -0.2 V. The calculated data in Table 1 show the effect of cathode vs. anode voltage on the activity ratio of K_3AlF_6 vs. $CeCl_3$. The low activity ratios at 0 and -0.3 V show that there are essentially no rare earth fluorides in the salt at cell voltages less negative than or equal to -0.3 V. Therefore, a low activity ratio in the molten electrolyte can be maintained until essentially all the aluminum has been removed from the spent fuel, and there will be no rare earth metal contamination of the aluminum product.

TABLE 1. Calculated Ratio of CeF_3 and K_3AlF_6 Activities as Function of Cathode vs. Anode Electrorefiner Voltage

cathode vs. anode (V)	0 V	-0.3 V	-0.4 V
$a_{CeF_3} / a_{K_3AlF_6}$	7.1×10^{-9}	6.9×10^{-2}	0.47

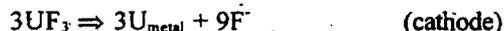
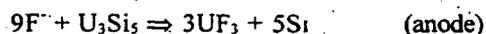
C. Consolidate Aluminum Dendrites

Because some salt will be adhering to the dendritic aluminum, the aluminum will be melted and coalesced into a single metal phase that melts at a temperature above the melting point of the salt. Then by cooling the melt below the melting point of aluminum, we can pour off the salt, which is still molten. The salt is then returned to the aluminum electrorefiner, and the consolidated aluminum ingot will be disposed of as low-level waste.

D. Uranium Electrorefining

After essentially all the aluminum has been removed from the spent fuel, all that remains in the anodes are the actinides, rare earths, and noble metals. The next step is to recover uranium from these anodes in a second electrorefiner. This uranium electrorefiner is identical to the one used for aluminum electrorefining, except that it is filled with a LiF-KF-UF₃ molten-salt electrolyte. Uranium electrodeposition in molten fluoride electrolytes has been reported and was found to be similar to electrodeposition in molten chloride electrolytes.

Because silicon was added in the initial head-end step, uranium will be present in the uranium electrorefiner anodes as a uranium silicide (e.g., U₃Si₅, USi₃). Therefore, the anode and cathode reactions, respectively, are:



As indicated by Figure 2, these reactions will occur at a cathode vs. anode voltage more negative than used for aluminum electrorefining. Calculations confirm this expectation. Whereas aluminum is electrorefined at -0.2 to -0.3 V, uranium is electrorefined at -0.4 to -0.5 V. After all the uranium has been extracted from the anodes, all that remains in the anodes are the noble metal fission products. Because the rare earth fluorides and TRU fluorides are more stable thermodynamically than UF₃, they will remain in the molten fluoride electrolyte.

E. Consolidate Uranium Dendrites

The uranium electrorefiner dendritic product is consolidated by a melting operation identical to the operation used to consolidate the aluminum dendrites obtained in the aluminum electrorefiner. The adhering salt is returned to the uranium electrorefiner.

F. Periodic Salt Scrubbing

With continued treatment of spent fuel, there will be a buildup of alkaline earth fluorides in the aluminum electrorefiner and of rare earth and TRU fluorides in the uranium electrorefiner. Eventually this buildup will result in an undesirable carryover into the electrorefiner product. It will be necessary to periodically scrub these metal fluorides from the salt or discard the salt. Salt scrubbing is the preferred choice because it will allow a single batch of salt to be used in each electrorefiner for the entire campaign. Several approaches are available for this periodic scrubbing, including chemical reduction and oxide precipitation.

IV. OXIDATION AND GLASS-FORMING STEPS

A. Oxidize Uranium Electrorefiner Anode Heels and Scrubbed Fission Products

The scrubbed alkaline earths, rare earths, and TRUs are then converted to oxides along with the metal that remains in the anode after uranium electrorefining. The operation is similar to an oxidation performed at ANL previously.³ The conversion is performed in an air oxidation furnace. The output from the furnace is an oxide powder with noble metal fines dispersed throughout the oxide.

B. Melt Fission Product Oxides and Fabricate Waste Glass

A small glass melter will be used to melt the oxide powder from the oxidation furnace, the compressed aluminosilicate trap, and additional glass formers. We anticipate that we will be able to formulate a glass within the specifications of the DWPF glass. To minimize waste volumes we have been careful throughout the process to add only glass-forming oxides (fibrous aluminosilicate) and silicon, which is converted to silica in the oxidation furnace. Alumina and silica are components of DWPF glass. The glass can then be poured into DWPF waste canisters, which will later be bundled into a waste package.

V. DEVELOPMENT STATUS

Many of the process steps in Figure 1 have already been successfully demonstrated. Uranium electrorefining in a high-throughput electrorefiner has been demonstrated at ANL. We have also successfully scrubbed rare earths out of a molten chloride salt by using chemical reduction

with lithium. Melting, casting, and consolidation steps were demonstrated in the Melt Refining Process at ANL. The key process operations that have yet to be demonstrated are electrorefining aluminum in a high-throughput electrorefiner and scrubbing the alkaline earth fission products out of the aluminum electrorefiner salt. Demonstrating these two steps is the main focus of our R&D effort for the next year. We are also beginning to design an engineering-scale high-throughput electrorefiner for demonstration with simulated spent fuel. Fabrication and installation of this electrorefiner are scheduled to be complete by this fall.

REFERENCES

1. D. Hampson, R. Frye, and J. Rizzie, "Melt Refining of EBR-II Fuel," in *Nuclear Metallurgy*, P. Chiotti (ed), volume 15, pp: 57-76 (1969).
2. E. Gay, W. Miller, and J. Laidler, "Proposed High Throughput Electrorefining Treatment of Spent N-Reactor Fuel," *Proceedings of Spring 1996 American Nuclear Society Meeting*, Reno, NV.
3. W. E. Miller, G. J. Bersteing, R. F. Malecha, M. A. Slawewski, R. C. Paul, and R. F. Fryer, "EBR-II Plant Equipment for Oxidation of Melt Refining Skulls", *Proc. 15th Conf. on Remote Systems Technology*, Chicago, IL, Am. Nucl. Soc., Hinsdale, IL, pp 43-51 (1967).

Chloride Volatility

Introduction

Chloride Volatility is based on completely volatilizing the fuels element and separating the gaseous constituents.

Process

The process consists of reacting the fuel with chlorine gas at high temperature (greater than 1200 C), which causes all the fuel constituents to form volatile chlorides. The gases are then separated by molten salt scrubbing and fractional condensation. There are four major unit operations:

- a) Chlorination and volatilization of all the fuel components at 1200 C,
- b) Removal of fission product-, transuranic-, nickel-, and chromium-chlorides in a zinc chloride scrubber at 400 C,
- c) Three condensers for removing, by fractional condensation at temperatures ranging from 164 to 2 C, $ZnCl_4$, $FeCl_3$, $ZrCl_3$, UCl_6 , $SnCl_4$, and I_2 vapors that pass through the scrubber, and
- d) Regeneration of the transferred spent molten salt by vacuum distillation to recover $ZnCl_2$, and $ZnCl_2$ -soluble $ZrCl_2$ for recycle, leaving the fission product-, transuranic-, nickel-, and chromium-chlorides as residue that would be converted to oxides or fluorides for vitrification.

Argon carrier gas and unreacted chlorine gas would be recycled, the Cl_2 content adjusted, and the steam split and passed through the unit operations in a continuous closed loop. Periodic shutdown of the coupled unit operations would occur for batch removal of fission product xenon and krypton gases from the carrier gas (such as cryogenic distillation), batch transfer of the molten salt to the molten salt regenerator, and batch removal of nonradiocative constituents and uranium from the condensers.

Waste Form

The small quantity of fission-product/transuranic-product HLW would be converted into a waste form for repository disposal. The conversion steps to a glass or glass/ceramic form could involve fluorination and then melting with glass frit additives, or conversion to oxides by heating at 1000 C with boric acid. Solvent extraction techniques, such as transuranic extraction (TRUEX), could be applied to the fission product and transuranic product chlorides for subsequent recycle.

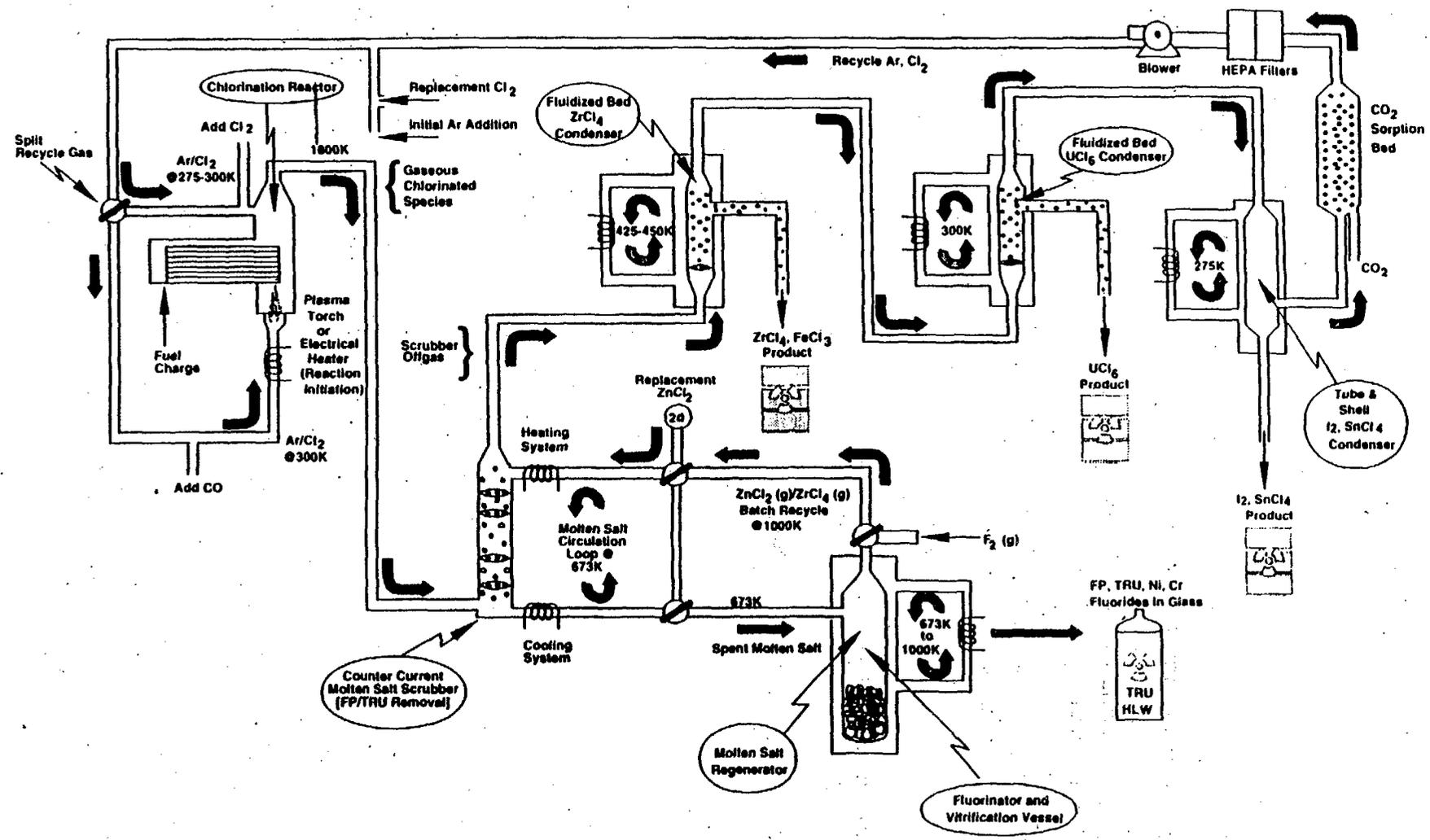
Advantages

- 1) A single waste form will be produced, saving some \$1.4 billion in repository costs over direct disposal.
- 2) Minimal secondary waste will be generated, compared with aqueous treatment.
- 3) Difficult separations will be possible that otherwise might not be achieved by aqueous treatment.
- 4) The treatment would substantially faster than aqueous treatment, saving substantial time and operation costs.
- 5) The fissile components will be removed, eliminating concerns and issues regarding criticality and theft of fissile material in a repository.

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F-17



Chloride Volatility Process.

Chop and Dilute

Introduction

The Chop and Dilute option is a mechanical volume compaction technique and is a relatively simple method of isotopic dilution of highly enriched uranium in aluminum-clad fuel.

Process

Chop and Dilute involves chopping the fuel elements into small pieces and mixing them with similar pieces of depleted uranium aluminum alloy. To preclude criticality in the repository, the resulting enrichment level of the mixture may have to be an equivalent low U-235 enrichment. This option has the significant disadvantage that it requires a larger number of waste canisters, inversely proportional to the enrichment levels allowed by the repository requirements. If the requirements for criticality control result in 1-2% U-235 enrichment, then the high level waste volumes will be greater than produced by traditional aqueous treatment methods.

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Can-In-Canister

Introduction

The Can-in-Canister option, which borrows its name from the plutonium immobilization option, would involve placing already canned highly enriched uranium spent fuel elements into DWPF canisters and then filling the canisters with high level waste glass. Criticality control combined with the space occupied by HLW glass would significantly limit the U-235 per canister. At three elements per canister, a total of 12,000 canisters would be required for the disposal of DOE aluminum-based SNF. The high level waste glass may be significantly cracked and would not provide a barrier against water.

A variation of the Can-in-Canister technique that combines the features of using the DWPF high level waste packages to dispose of fuel, while allowing a wide variety of potential treatments, is co-disposal.

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