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FINAL ENVIRONMENTAL IMPACT STATEMENT

on a

Proposed Nuclear Weapons Nonproliferation

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Description and Impacts of Storage Technology Alternatives

Summary

This appendix presents a description and evaluation of currently available spent nuclear fuel storage technologies, and their applicability to foreign research reactor spent nuclear fuel. These technologies represent the range of alternatives that would be available to implement the proposed action. Some of these technologies are currently in use at U.S. Department of Energy (DOE) facilities. Several dry storage cask and/or building designs have been licensed by the U.S. Nuclear Regulatory Commission (NRC) and are operational with commercial nuclear power plant spent fuel at several locations.

This appendix also discusses potential storage sites and impacts of foreign research reactor spent nuclear fuel storage at these locations. The major sections in this appendix are:

- Section F.1 Description of Existing and Proposed Technologies for Storage of Spent Nuclear Fuel
- Section F.2 Storage Technology Evaluation Methodology
- Section F.3 Selection of Storage Technologies for Further Evaluation
- Section F.4 Environmental Impacts at Foreign Research Reactor Spent Nuclear Fuel Management Sites
- Section F.5 Occupational Radiation Impacts from Receipt and Handling of Foreign Research Reactor Spent Nuclear Fuel
- Section F.6 Evaluation Methodologies and Assumptions for Incident-Free Operations and Hypothetical Accidents at Management Sites
- Section F.7 Economic Evaluation of Foreign Research Reactor Spent Nuclear Fuel Storage and Related Management Alternatives

Figure F-1 presents the different spent nuclear fuel storage technologies, which are divided into wet and dry systems and further classified by their materials of construction (i.e., concrete, metal), location (i.e., aboveground or belowground) and size (i.e., cask versus vault building or pool). The final level of

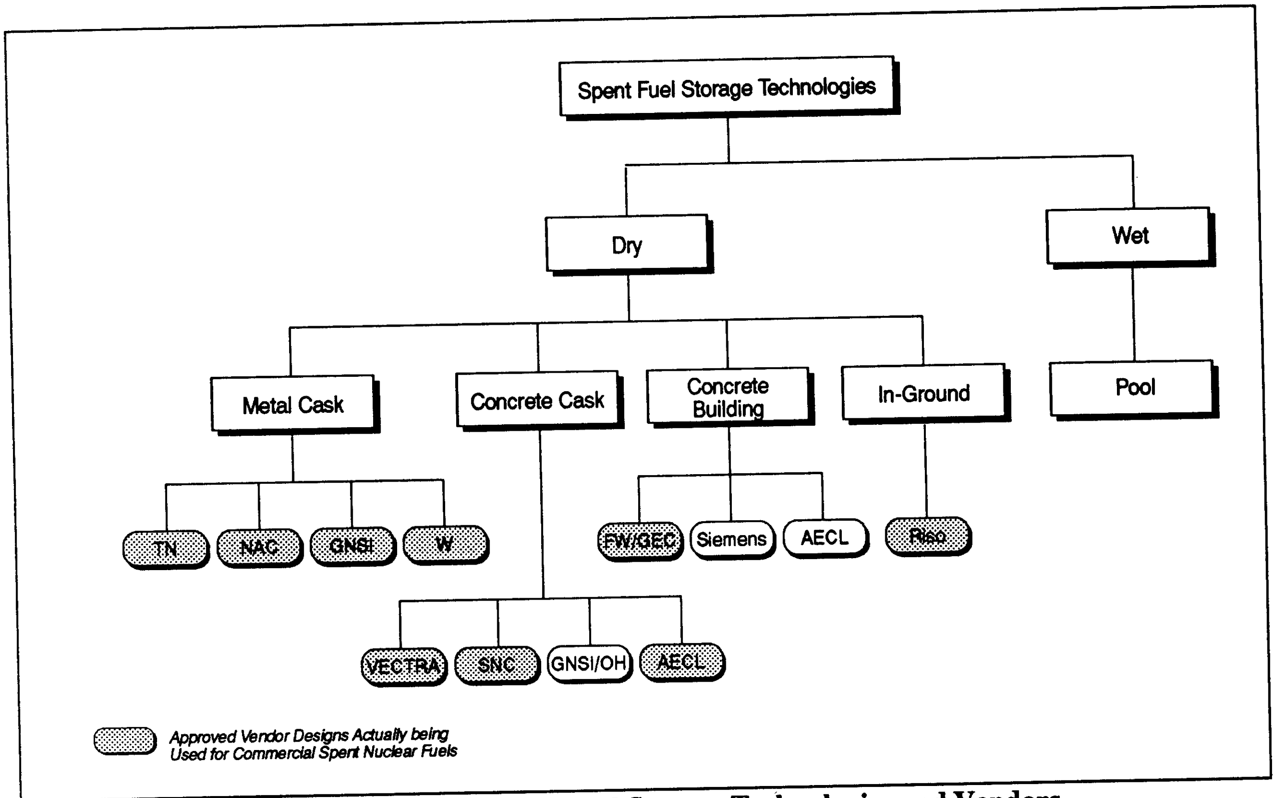


Figure F-1 Spent Nuclear Fuel Storage Technologies and Vendors

nuclear fuel with only minor, easily implemented modifications, such as interior baskets for holding the spent nuclear fuel. Use of existing facilities at a site for staging and characterization favors a cask storage approach, while a stand-alone, separate spent nuclear fuel storage approach requires a vault and other support facilities. Schedule and monetary considerations favor casks over the vault for sites with existing facilities, and this is why most domestic utilities are pursuing dry casks for long-term storage of spent nuclear fuel. Casks are the only independent spent nuclear fuel storage installation designs that have received certification by the NRC in accordance with 10 Code of Federal Regulations (CFR) 72 Appendix K.

The evaluation indicates that both wet and dry storage of foreign research reactor spent nuclear fuel appear acceptable for the time periods envisioned for the proposed action (i.e., through 2036). Commercial spent nuclear fuel dry storage systems require a minimum wet pool storage time or cooldown period of approximately 5 years after discharge from the nuclear reactor prior to emplacement into dry storage. In actual practice, this usually averages around an 8-year average cooldown period and, frequently, the commercial spent nuclear fuel has had over a 10-year cooldown period in a wet pool prior to emplacement into dry storage. This cooldown period ensures that licensed conditions for cladding temperatures (based upon potential corrosion, and usually around 350°C, or 630°F) are not exceeded. Foreign research reactor spent nuclear fuel has a lower cladding temperature limit based upon a phase transition in the aluminum metal cladding; this aluminum cladding limit has been identified as 175°C (315°F). Thus, a maximum research reactor spent nuclear fuel, without any clear preference. It should be noted that a research and development project to examine the applicability of aluminum-clad spent nuclear fuel dry storage at the Savannah River Site was initiated in Fiscal Year (FY) 1994.

The utilization of dry storage methods for foreign research reactor spent nuclear fuel requires the acquisition of racks, baskets, storage canisters, and/or casks. New construction would be required for dry vaults, except for several existing facilities at the Nevada Test Site and Idaho National Engineering Laboratory.

The utilization of wet storage methods requires a lined basin within a seismically qualified facility with the ability to maintain water chemistry and handle liquid radioactive waste. Currently, there are few existing DOE facilities in this category, and none have sufficient capacity to accommodate all of the foreign research reactor spent nuclear fuel. Thus, the selection of wet storage would require DOE acquisition of a facility, either by new construction or purchase of an existing facility such as the Barnwell Nuclear Fuels Plant (BNFP) that is owned by Allied General Nuclear Services. A summary of storage technology characteristics is given Table F-1. Sections F.4, F.5, and F.6 address environmental impacts, occupational dose, and accident consequences for storage. Section F.7 discusses costs in detail.

Table F-1 Summary of Storage Technology Characteristics for Commercial and Foreign Research Reactor Spent Nuclear Fuel

Storage Technology	DOE Site Status (New or Existing)	Land Use Ha (Ac)	Annual Low-level Waste (m^3)[a]	Potential Annual Spent Nuclear Fuel Storage Public Impact (LCFs)	Lead Time until Spent Nuclear Fuel Storage (Years)
Dry Vault - Utility Fuel	New	4 (5)	1-4	NA	2-3
Dry Cask (Concrete) - Utility Fuel	New	4 (5)	1-4	NA	2-3
Wet Pool - Utility Fuel	New	2 (3)	1-4	NA	3-5
Dry Vault - Savannah River Site Research Reactor Fuel	New	4 (8)	16	0	5-10[b]
Dry Cask - Savannah River Site Research Reactor Fuel	New	4 (5)	16	0	3-5[b]
Wet Pool - Idaho National Engineering Laboratory Research Reactor Fuel	New	2 (3)	12	2.4×10^{-12} to 2.5×10^{-10}	5-10[b]

NA = Not Available; LCF = Latent Cancer Fatality

[a] *Low-Level Waste generation decreases significantly if spent nuclear fuel is only being stored, without additional spent nuclear fuel receipts. To convert to ft^3, multiply by 35.3.*

[b] *To allow for extended periodic examination and characterization of fuel.*

DOE currently has pilot-scale experience with dry storage of spent nuclear fuel, and there are no identified technical constraints that would prevent dry storage of foreign research reactor spent nuclear fuel. There would be some need, however, for characterization, canning, and periodic inspection and monitoring. Both NRC-licensed and not yet licensed dry storage designs are readily available from commercial vendors. NRC-licensed designs have the following advantages:

- specific NRC requirements have been met that are equivalent to DOE requirements and guidance,
- extensive, interactive technical safety reviews have already been conducted between the supplier and the regulator,
- peer and public review has occurred as part of the licensing process,
- proven applications are in operation at commercial nuclear power plant sites, and
- documentation and quality assurance requirements have been satisfied.

For sites with an existing spent nuclear fuel infrastructure that includes facilities for spent nuclear fuel receipt, examination, and loading, a modular approach based upon casks can be implemented rapidly to meet Phase 1 requirements using standard funding and procurement capital appropriation methods. The casks could also be used for Phase 2, and their usage would avoid additional procurement. A modular dry vault approach represents an integrated self-contained, stand-alone facility, and can be used at any of the proposed management sites. However, construction of the vault could represent a major project or major

At the level of accuracy in the costs presented here, alternatives based on chemical separation of

… loaded onto a truck and transferred to a previously constructed

F.1.1.1.3 Aboveground Free-Standing Dry Storage Building (Vault)

Vault storage consists of a large concrete aboveground building enclosing a vertical or horizontal array of spent nuclear fuel storage metal tubes and support systems, The advantages for the vault type of dry storage, as compared to all other storage technologies, are the following. For large quantities of spent nuclear fuel assemblies, the vault may have economic advantages when compared with either type of cask system. The heat removal is passive. The heat removal capacity for a properly designed vault is large, and therefore, there should be little concern for thermal limits being imposed (although there may be individual fuel decay heat limits). The vault which is licensed in the United States and abroad, has no high temperature limit associated with concrete. However, there is a low temperature limit because the secondary fuel confinement barrier is ferritic steel. To comply with current NRC 10 CFR 72 regulations, all spent nuclear fuel storage systems must have two confinement barriers. The intact fuel cladding is considered the first confinement barrier, and the cask or vessel is considered the secondary confinement barrier. The vault has a major advantage over all other types of dry storage because it provides a shielded means for loading the spent nuclear fuel on the vault premises. Another important advantage of the vault is the ease of spent nuclear fuel retrieval and monitoring while in storage. The vault includes facilities for inspection, placement in containers, and drying of wet fuel. The weight/volume of stored fuel is not a limiting factor. This type of system is currently in use at Fort St. Vrain in Colorado, at Wylfa, Wales, and [illegible] is under construction at the PAKs nuclear power plant in Hungary.

The disadvantage is that, for small quantities of spent nuclear fuel, the cost may be higher than either the metal or concrete cask systems since a vault requires greater capital outlays.

F.1.1.1.4 Inground Lined and Unlined Wells With or Without Casks

The RISO National Laboratory's inground concrete block design relies on forced air convection heat transfer from the existing handling bay ventilation system, which includes High Efficiency Particulate Air filters and an air humidity monitoring system. Forced air heat removal is accomplished by directing the air around spent nuclear fuel containers and out through tubes embedded in the concrete. Like the pool storage systems, the RISO National Laboratory's system relies on active heat removal systems.

F.1.1.1.5 Hot Cell Facilities

Although hot cells are available at many facilities, including the Savannah River Site, the Idaho National Engineering Laboratory, and the Nevada Test Site, they can best be considered for small quantities of transportation. For commercial utilities, this implies satisfaction of 10 CFR 71 requirements for transportation and 10 CFR 72 requirements for storage. By minimizing fuel handling operations, the dose for workers can be reduced, and the number of additional low-level waste products can be reduced. Minimization of fuel handling may also result in cost reductions, although this case has not been made. For a multi-purpose cask, satisfaction of 10 CFR 60 requirements is also necessary. DOE's Office of Civilian Radioactive Waste Management was actively pursuing a program to develop multi-purpose canister for domestic use (EG&G, 1994b; DOE, 1994f; DOE, 1994b; DOE, 1994c). However, DOE has decided in November 1995 to withdraw its proposal to prepare the EIS for this project.

The dual-purpose cask systems that are currently proposed offer a reduction in handling of the spent nuclear fuel in the storage to transportation operations, and multi-purpose cask systems offer an even greater reduction in handling in the operations involved at the repository site. However, no detailed cost/benefit analyses have been undertaken for foreign research reactor spent nuclear fuel. Furthermore, there is no basis at this time for concluding that either the "waste form" (intact spent nuclear fuel assemblies) or the sealed container will be compatible with the repository requirements. It is premature to draw any conclusion on the desirability to proceed with a multi-purpose cask system for foreign research reactor spent nuclear fuel use.

F.1.1.2 Specific Dry Storage Designs

There are no currently licensed dry storage systems specifically for foreign research reactor spent nuclear fuel in the United States. There are, however, many examples of dry spent nuclear fuel systems licensed by the NRC for commercial fuel. Table F-2 provides an overview of current manufacturers of dry storage systems. Table F-3 is a listing of dry storage systems currently licensed in the United States.

Dry storage systems must meet many design criteria, such as protection of fuel from degradation, shielding, thermal, criticality safety, structural integrity of confinement vessel, structural integrity of shielding, mechanical handling of fuel assemblies or canisters, containment and operational aspects. Some of these criteria are interrelated. For example, thermal criteria are designed to maintain fuel and cladding structural integrity. Shielding, thermal, and criticality parameters are the most important and are discussed in the following sections.

F.1.1.2.1 Shielding Design Comparisons

A spent nuclear fuel storage system must provide for adequate shielding of both the gamma and neutron radiation that emanate from irradiated nuclear fuel. The shielding must be designed to reduce the combined gamma and neutron dose rate to values that are below the limits for the public at the site boundary, collocated workers, and workers at the fuel storage facility. These limits are determined by Federal regulations such as 10 CFR 72 and 10 CFR 20. Shielding is designed for the maximum expected gamma and neutron source term, which is determined by performing computer code analyses of the nuclear fuel that account for the initial fuel fissile material inventory, its burnup in the reactor core, and the time after removal from the reactor (i.e., decay time) prior to its anticipated placement in the storage facility. The selection of a bounding and conservative set of these parameters results in the calculation of the highest possible gamma and neutron source term to be used in shielding design and analyses.

Shielding for gamma radiation relies on the use of high atomic weight or density materials, which attenuate and absorb gamma rays. The material selection depends on design limitations regarding shield thickness, cost, strength, and weight. The five materials which are almost always used in spent nuclear fuel storage facilities for gamma shielding are water, lead, steel, ductile iron or concrete. Lead and steel, having much higher densities and atomic weights than concrete and water, can provide relatively more

Table F-2 Dry Storage Technology Systems

resin, and polyethylene. As in the case for gamma shielding, design factors in material selection include cost, density, weight, and safety.

The design of spent nuclear fuel storage facility shielding must also incorporate other factors along with cost, density, weight, and safety. Shielding usually performs a second function as a heat transfer medium from the spent nuclear fuel decay heat to the environment, and must therefore be able to effectively remove heat without exceeding fuel and shielding storage temperature limits. In some instances, the shielding also performs a structural function, either in handling or support.

Table F-4 shows a comparison of specific designs with a view toward shielding considerations. All of these designs will be discussed in more detail in subsequent sections of this appendix.

F.1.1.2.2 Thermal Design Comparisons

Spent nuclear fuel storage facilities are designed to effectively remove spent nuclear fuel decay heat during both incident-free operation and postulated accident conditions. Thermal design limits include long-term fuel storage cladding temperature to maintain cladding integrity and, in some cases, temperature limits of structural and/or shielding materials. Unlike pool storage systems, most of the dry storage systems emphasize passive heat removal. In contrast, active systems in wet pools include pumps, make-up water systems, filtration and water treatment systems, and heat exchangers.

All dry storage designs encapsulate the fuel, after it is dried, in a metal canister or tube that is evacuated (vacuum dried) and then filled with an inert gas such as helium. Helium is frequently used for its relatively high thermal conductivity that enhances heat conduction and heat transfer from the fuel to the encapsulating metal canister. Helium's inert properties also inhibit cladding corrosion. Since all the dry fuel storage technologies utilize a metal canister to enclose spent nuclear fuel, the first modes of heat transfer from the fuel to this canister's walls are heat conduction and radiation from the fuel cladding surface through the inert gas to the inside wall of the metal canister. Decay heat transfer from the encapsulating canister to the environment is accomplished by several different mechanisms dependent upon the specific storage design technology.

The dry metal cask design relies on its solid thick metal wall for conduction heat transfer from the fuel storage cavity to the atmosphere. Metal cask conduction heat transfer is not susceptible to any accident or degradation. This thermal design is inherently easy to analyze because conduction is a well-known heat transfer mechanism, and the thermal conductivity of such metal cask materials as carbon steel, stainless steel, and ductile cast iron is well known over the range of temperatures and conditions that are expected in the cask while storing spent nuclear fuel. With known design fuel decay heat, cask geometry (i.e., cask wall thickness), conduction material composition, and suitably conservative heat transfer assumptions from the cask metal surface to the ambient air, the temperature distribution within the cask and maximum fuel cladding temperature can be calculated with a high degree of certainty.

The dry concrete cask design uses a combination of conduction, natural convection, and radiation heat transfer to remove decay heat from the stored spent nuclear fuel and maintain acceptable operating temperatures. An air passageway around the storage canister is provided in this design because the relatively low thermal conductivity and allowable operating temperature limit of concrete, as compared to metal, prevent the concrete shield walls from serving as the primary means of decay heat removal. Radiation streaming requires that the inlet and outlet air passages to the cavity surrounding the canister be designed as a geometric labyrinth with suitable bends. One concrete cask design, the Atomic Energy of Canada, Ltd. SILO, does not have air passages but instead relies solely on conduction through solid

Table F-4 Comparison of Shield Design Parameters for Spent Nuclear Fuel Dry Storage Systems Currently Licensed in the United States

Manufacturer	Model	Shield Material	Shield Thickness	Design Limit Surface Dose Rate[a]
Nuclear Assurance Corporation	S/T	S.S., Lead, NS4FR	Radial: 20.3 cm (8 in) S.S. & 17.8 cm (7 in) NS4FR Axial: 12.7 cm (5 in) S.S.-PB & 7.6 cm (3 in) NS4FR	1 milliSievert/hr (100 mrem/hr)
Transnuclear, Inc.	TN-24	Borated Resin, C.S.	NA	Side: 0.57 milliSievert/hr (57 mrem/hr) Top: 0.11 milliSievert/hr (11 mrem/hr) Bottom: 0.45 milliSievert/hr (45 mrem/hr)
	TN-40	Borated Resin, C.S.	Radial: 21.6 cm (8.5 in) C.S. 11.4 cm (4.5 in) Resin Bottom: 22.2 cm (8.75 in) C.S. Top: 15.9 cm (6.25 in) Cast Iron	Side: 0.58 milliSievert/hr (58 mrem/hr) Top: 0.26 milliSievert/hr (26 mrem/hr) Bottom: 12.75 milliSievert/hr (1,275 mrem/hr)
Westinghouse Electric Corporation	MC-10	NS-3, C.S.	Radial: 25.4 cm (10 in) Steel, 7.6 cm (3 in) NS-3 Bottom: 25.4 cm (10 in) steel	2 milliSievert/hr (200 mrem/hr)
General Nuclear Systems, Inc.	CASTOR V21	Cast Iron, S.S., Polyethylene Rods	Radial: 30.5 cm (12 in) Bottom: 27.9 cm (11 in) Top: 39.1 cm (15.4 in) Rods Radial: 72-6.1 cm (2.4 in) Diameter	2 milliSievert/hr (200 mrem/hr)
VECTRA	NUHOMS 7P, 24P, and 52B	Concrete, S.S.	Side: 45.7/60.1 cm (18/24 in) Rear: 60.1 cm (24 in) Roof: 91.4 cm (36 in)	2 milliSievert/hr (200 mrem/hr) (at air inlet)
Sierra Nuclear Corporation	Ventilated Storage Cask-24	Concrete RX-277, Hydrogenated Concrete, C.S.	Radial: Steel & Concrete Top: RX-277 & Steel Bottom: Steel & Concrete	Side: 0.20 milliSievert/hr (20 mrem/hr) Top: 0.50 milliSievert/hr (50 mrem/hr) Air Inlet or Outlet: 1 milliSievert/hr (100 mrem/hr)
FW/GEC	MDV	Concrete	106.7 cm (42 in)	0.21 milliSievert/hr (21 mrem/hr)

NA = Not Available; C.S. = Carbon Steel; Pb = Lead; S.S. = Stainless Steel; NS-3 = Concrete; NS4FR = Special Fire-Resistant Castable Resin; RX-277 = Special Concrete with Extra Hydrogen; FW/GEC = Foster Wheeler/GEC Alsthom Engineering Systems, Ltd. (United Kingdom); MDV = Modular Dry Vault

[a] *These are limits established for commercial spent nuclear fuel assemblies. The dose rate expected from storage of foreign research reactor spent nuclear fuel is likely to be lower.*

concrete. The SILO's thermal design is acceptable only because it is limited to a much smaller total decay heat power than the air passage concrete casks.

Since a passive design is an underlying requirement of all dry concrete cask designs, the total airflow path from the cask air inlet to its outlet must include a sufficient elevation change to ensure natural convection airflow under all expected meteorological and heat load conditions.

The heat transfer from the canister follows two parallel paths: (1) convection from the surface of the canister to the naturally-induced airflow through the canister cavity, and (2) radiation and conduction heat transfer from the canister across the air in the cavity to the concrete shield and then conduction through the concrete shield wall thickness to the ambient air outside the concrete. Natural convection air heat removal is greater than the radiation and conduction through the air layer and concrete shield.

The heat transfer design of the concrete cask is vulnerable to accidents in which significant blockage of the air inlets and/or outlets restricts or prevents sufficient airflow into the canister cavity. Multiple inlets and outlets at different, and sometimes diametrically opposed, locations around the cask are used to reduce the likelihood of such an accident. Conservative adiabatic heatup analyses for these designs with commercial spent nuclear fuel have shown that temperature limits are not approached in more than 24 hours, even if the airflow inlets and outlets are completely blocked. Therefore, concrete cask sites have included a daily visual surveillance frequency for inspection of air inlets and outlets to ensure that they are not blocked. The adiabatic heatup for foreign research reactor spent nuclear fuel and its concomitant surveillance frequency may be different.

The concrete cask thermal design also requires more complex analyses for temperature distribution in both the fuel and the concrete due to the complex multidimensional and combined conduction-radiation-convection modes of heat transfer. An important thermal design issue for the concrete casks is proof that natural convection buoyancy-driven airflow will be induced through the inlet-cavity-outlet path under the entire range of expected wind and decay heat conditions, including the possibility of partial blockage that may be obscured from outside visual inspections. Unlike metal casks, which only have the fuel cladding temperature as a thermal limit, concrete casks are also limited by both the absolute magnitude and gradients of temperature within the concrete.

The concrete vault storage building represents a larger version of the concrete cask design in the realm of heat transfer. An array of vertically or horizontally oriented metal canisters enclosing spent nuclear fuel is surrounded by a concrete building with labyrinth air inlet and outlet passages. With the exception of size, this design utilizes the same modes of heat transfer as the concrete cask. Its inherently larger flow areas for inlets and outlets and typically larger elevation from inlet to outlet provide a greater natural convection airflow and reduce vulnerability to airflow passage blockage.

Specific Thermal Features

Thermal design performance parameters of specific manufacturers' dry storage technologies are presented in Table F-5. This table shows that all dry spent nuclear fuel storage technologies use radiation and conduction as heat transfer mechanisms, and that concrete-based systems also rely on internal air passage natural convection heat transfer. All the systems have fuel cladding temperature limits, but systems relying on concrete also have concrete temperature limits.

Pool storage systems utilize an active cooling system with pumps and heat exchangers that remove decay heat transferred to the pool water from stored fuel via conduction and natural convection. The relatively large mass and heat capacity of the pool water provide a significant margin of time before the pool water reaches its boiling temperature in the event of a cooling system failure.

The RISO National Laboratory's inground concrete block design relies on forced air convection heat transfer from the existing handling bay ventilation system, which includes High Efficiency Particulate Air filters and an air humidity monitoring system. Forced-air heat removal is accomplished by directing the heating, ventilation, and air conditioning air around the stored fuel, and then out through separate tubes embedded in the concrete. Like the pool, the RISO National Laboratory's system relies on active heat removal systems.

The Atomic Energy of Canada, Ltd., SILO is an exception to the previously discussed concrete cask designs because it relies solely on conduction through a solid concrete structure for decay heat removal,

Table F-5 Comparison of Thermal Design Parameters for Spent Nuclear Fuel Dry

Suitable criteria for establishing nuclear criticality safety have been documented (ANSI, 1984b, 1983, and 1975b). These documents deal specifically with, respectively, the storage of commercial spent nuclear fuel outside of the reactor and in dry storage installations.

Another conservative aspect of these criticality analyses is the requirement that a sensitivity study be

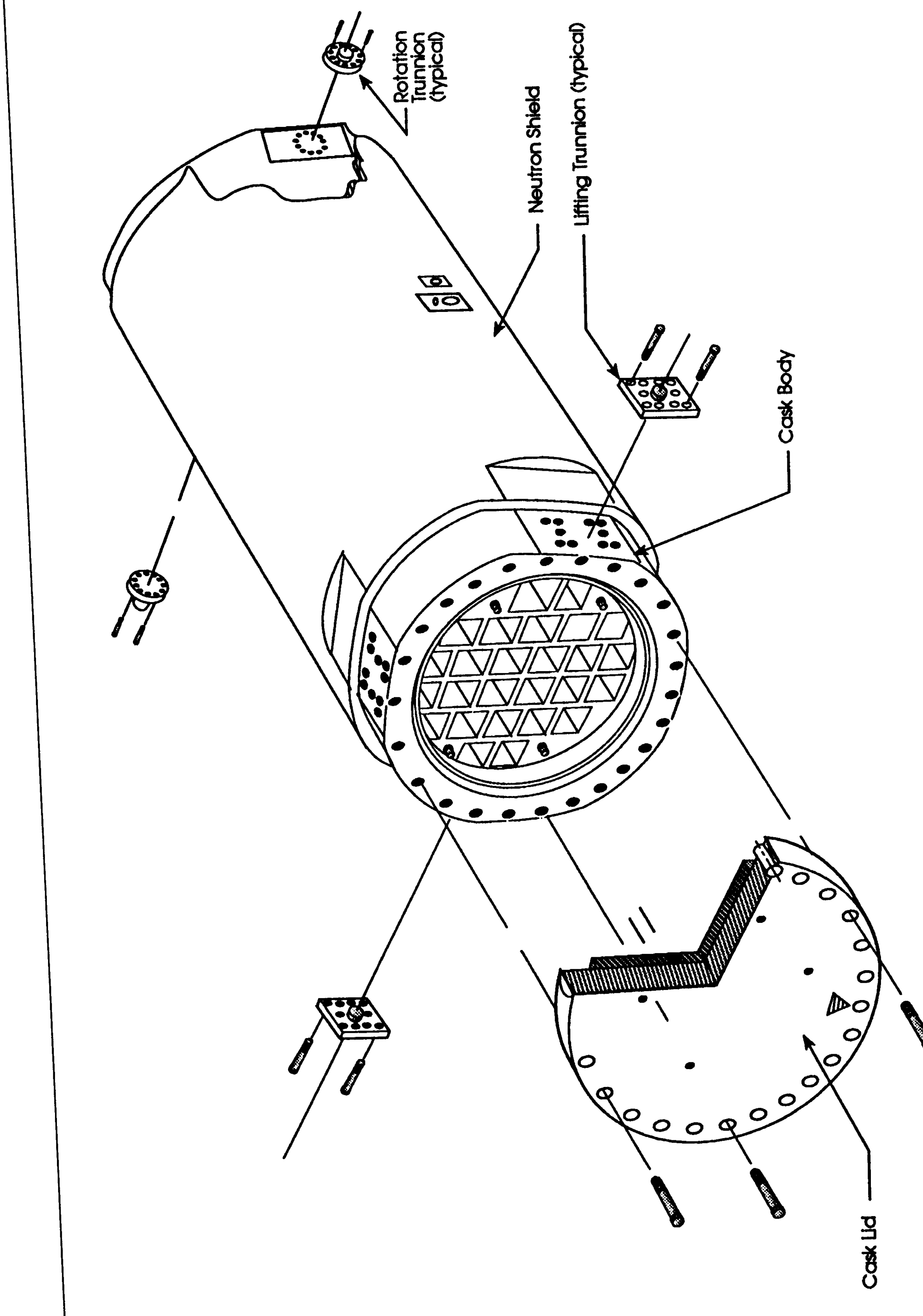


Figure F-2 Nuclear Assurance Corporation Spent Nuclear Fuel Storage Cask Body NAC S/T

NRC Certification or Basis for License

The NRC has granted a Certificate of Compliance to Model Nuclear Assurance Corporation S/T (Certificate Number 1002). Nuclear Assurance Corporation Model NAC-C28 S/T is also certified with Certificate Number 1003. The basis for these certificates is 10 CFR 72 Subparts K and L. Nuclear Assurance Corporation model NAC-I28 is currently licensed on a site-specific basis at Surry Nuclear Power Plant based on 10 CFR 72 Subparts A through I.

F.1.1.2.4.2 General Nuclear Systems, Inc. CASTOR V/21

Description of General Nuclear Systems, Inc. CASTOR V/21

In the United States, the General Nuclear Systems, Inc. CASTOR V/21 has been approved by the NRC and is in use at the Surry Nuclear Power Plant. This design relies on thick ductile cast iron and polyethylene as both its gamma and neutron shields. Ductile cast iron contains significant quantities of nodular graphite, which is essentially carbon, a good neutron shield. Polyethylene is a form of plastic that is high in hydrogen. The ductile cast iron shield is 30.5 cm (12 in) thick. Additional neutron shielding is provided by seventy-two 6.1 cm (2.4 in) diameter polyethylene rods placed in axial holes in the cast iron wall. The top lid shielding is 39.1 cm (15.4 in) of stainless steel, and the bottom lid shielding is 27.9 cm (11 in) of ductile cast iron. The V/21, holding 21 Pressurized Water Reactor fuel assemblies at Surry, weighs 96 metric tons (106 tons) fully loaded. A sketch of the CASTOR V/21 is presented in Figure F-3. The shielding design basis is for a surface contact dose rate less than 200 mrem/hr. There is a wide range of CASTOR designs for a variety of fuel types, including test reactor fuel. A conceptual design [CASTOR Material Test Reactor (MTR) 2] for a dual-purpose, transport/storage cask for research reactor fuel has been developed. This cask uses the same basic ductile cast iron body for shielding.

NRC Certification or Basis for License

The NRC has granted Certificate of Compliance Number 1000 for the General Nuclear Systems, Inc. model CASTOR V/21 under the terms of 10 CFR 72 Subparts L and K (Models X/28 and X/33 are not currently licensed, but are being reviewed by the NRC).

F.1.1.2.4.3 Westinghouse Electric Corporation MC-10

Description of Westinghouse Electric Corporation MC-10

The Westinghouse Electric Corporation MC-10 metal cask has been approved by the NRC and is in use at the Surry Nuclear Power Plant site (NRC, 1987). This cask design utilizes thick carbon steel and BISCO NS-3 hydrogenated concrete for shielding. The NS-3 provides neutron shielding, while the carbon steel is used for gamma shielding. Total radial neutron and gamma shielding is approximately 33 cm (13 in), while axial shielding is about 25.4 cm (10 in). The design surface contact dose rate is 200 mrem/hr, which bounds the actual vendor-calculated maximum surface contact dose rates of 7, 38, and 57 mrem/hr at the top, side, and bottom of the cask. The MC-10 was designed to hold 24 Pressurized Water Reactor fuel assemblies and weighs 103 metric tons (113.3 tons) fully loaded.

NRC Certification or Basis for License

The NRC has issued Certificate of Compliance Number 1001 for the metal cask model MC-10 in accordance with the terms of 10 CFR 72 Subparts L and K.

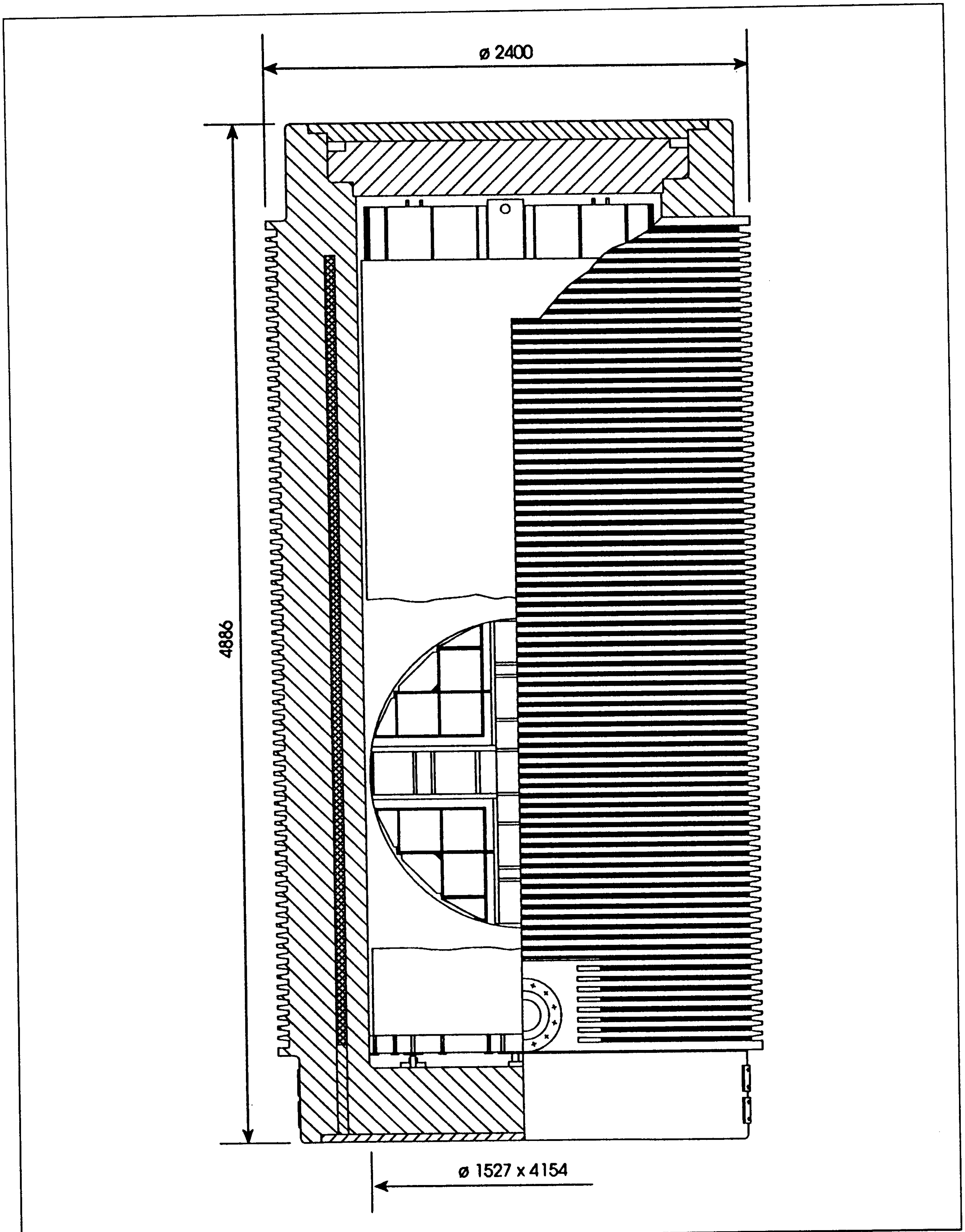


Figure F-3 The CASTOR V/21

F.1.1.2.4.4 Transnuclear, Inc. TN-24 and TN-40

Description of Transnuclear, Inc. TN-24, TN-40

The Transnuclear, Inc. design has been developed and produced for a large number of storage and transportation systems for radioactive materials, including spent nuclear fuel. The TN-24 and TN-40 models store 24 and 40 spent Pressurized Water Reactor fuel assemblies, respectively. TN systems feature metal casks for both transportation and storage of spent fuel. The TN-24 is an NRC-licensed storage cask that uses carbon steel for gamma shielding and a borated resin for neutron shielding (NRC, 1989). The TN-40 is a newer model that uses a two-metal shell design, with the inner shell consisting of high quality carbon steel for containment and the outer shell providing shielding and heat transfer, but of a lower quality steel. For the two models, top and side contact dose rate limits are less than 100 mrem per hour, but the bottom of the cask may have a contact dose rate limit as high as 1,275 mrem/hr. It should be noted that the normal configuration for these casks is to be standing upright on their bottoms, thereby precluding exposure to this relatively higher dose rate. A sketch of a Transnuclear, Inc. TN cask is shown in Figure F-4.

NRC Certification or Basis for License

The TN-24 model has been issued NRC Certificate of Compliance Number 1005 and is licensed according to 10 CFR 72 Subparts L and K. The TN-40 model is licensed on a site-specific basis at the Prairie Island Nuclear Power Plant in Minnesota (owned by Northern States Power) under the provisions of 10 CFR 72 Subparts A through I. The Transnuclear, Inc. Model TN-32 is not yet approved.

F.1.1.2.4.5 VECTRA Design NUHOMS-7P, -24P, and -52B

Description of VECTRA NUHOMS-7P, -24P, and -52B

VECTRA's NUHOMS designs utilize a horizontal concrete dry storage system for spent nuclear fuel (NUTECH, 1988). The NUHOMS-7P and NUHOMS-24P designs have been approved by the NRC for Pressurized Water Reactor spent nuclear fuel and are in use at the Robinson, Oconee, and Calvert Cliffs Nuclear Power Plant sites. The NRC approved the use of NUHOMS-52B for the Brunswick power plant, but the utility shipped this spent nuclear fuel to its Robinson plant. The NUHOMS design uses concrete as both gamma and neutron shielding. The requirement for internal air passages to allow natural convection heat removal from the metal storage canister placed within the concrete structure required 90 degree bends in the concrete shield for air passages to avoid radiation streaming and more detailed shielding analyses. The reinforced side wall concrete shield thickness is 45.7 or 61 cm (18 or 24 in) depending on location in the array, while the rear wall is 61 cm (24 in) thick and the roof is 91.4 cm (36 in) thick. The maximum surface contact dose rate limit at the air inlet is 200 mrem/hr. A sketch of the NUHOMS-24P system is shown in Figure F-5.

NRC Certification or Basis for License

NUHOMS models 7P and 24P are licensed at specific sites under the provisions of 10 CFR 72. Vectra has also received a license from the NRC for their standardized NUHOMS-24P and -52B models for use by the light water reactor utilities.

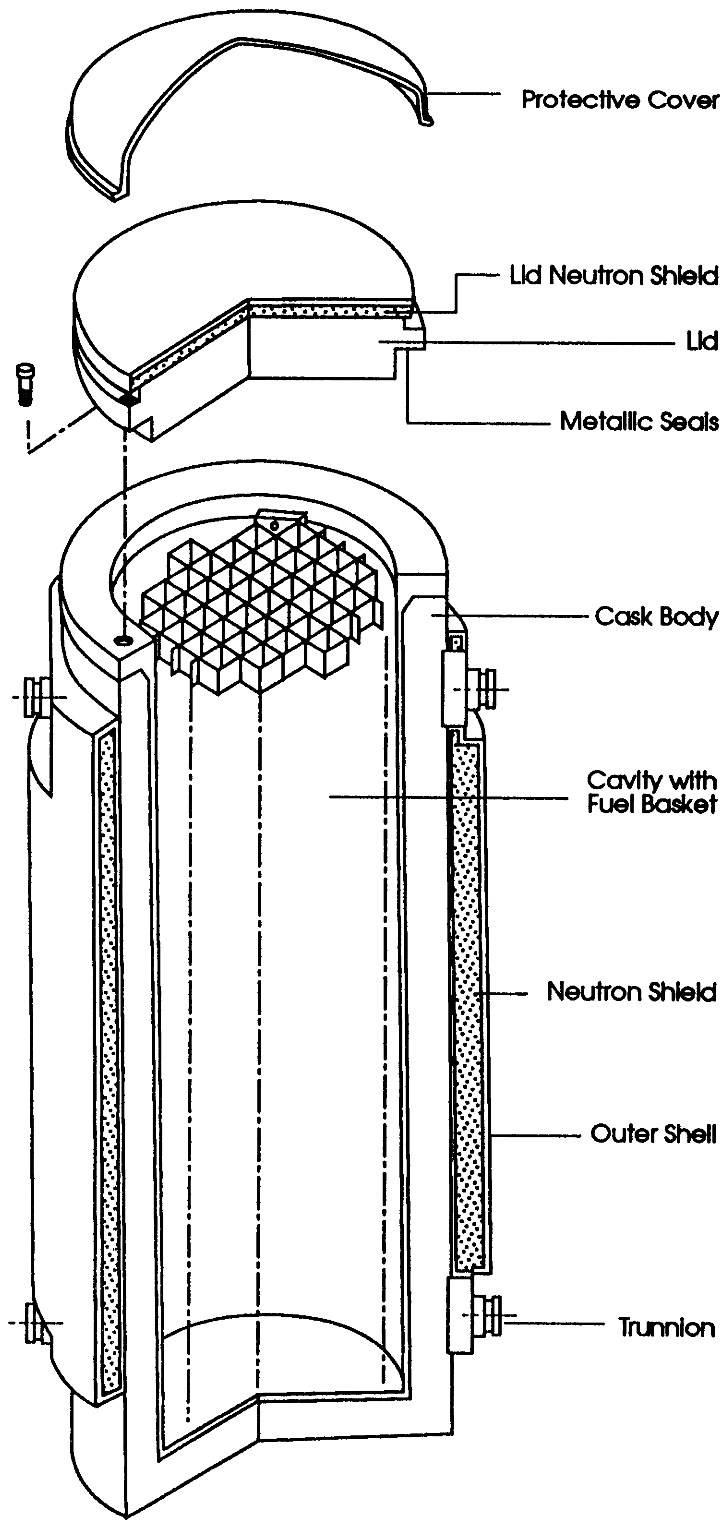
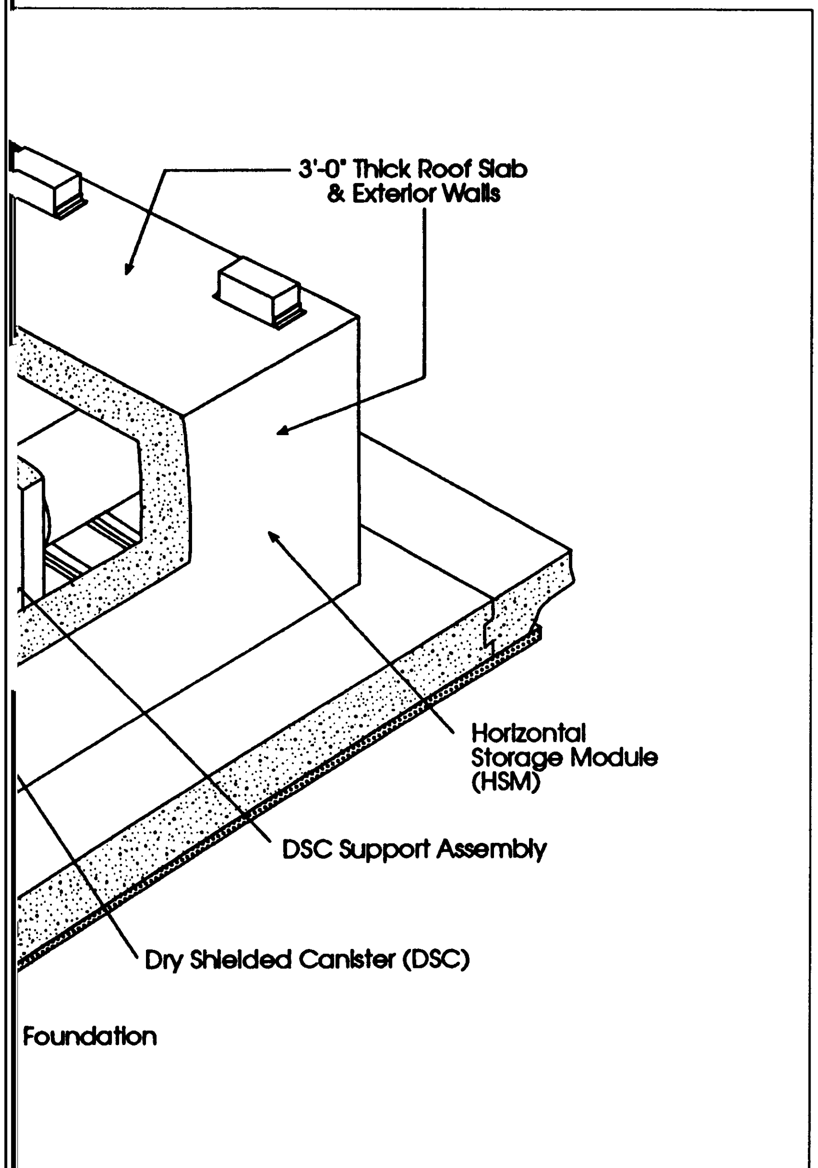


Figure F-4 The Transnuclear, Inc. TN Cask



l Storage Module Components

F.1.1.2.4.6 Modular Dry Vault

Description of Modular Dry Vault

The modular dry vault spent nuclear fuel storage system [designed by Foster Wheeler/GEC Alsthom Engineering Systems, Ltd. (United Kingdom)] is the only vault system in the United States that has been approved by the NRC and is in operation at the Fort St. Vrain nuclear power plant site. The modular dry vault places spent nuclear fuel in vertically oriented cylindrical steel fuel storage containers which are then inserted into a steel charge face structure within the thick concrete structure. A labyrinth airflow passage system provides natural convection airflow for decay heat removal. The shielding is provided by the 106.7 cm (42 in) thick concrete walls and the labyrinth airflow passages. For the Fort St. Vrain fuel, maximum design modular dry vault surface dose rate is 21 mrem/hr. A picture of the cross section of the modular dry vault is shown in Figure F-6.

NRC Certification or Basis for License

The modular dry vault model has been approved by the NRC for the site-specific application at Fort St. Vrain. The basis for the license is 10 CFR 72.

F.1.1.2.4.7 Ventilated Storage Cask System (VSC-24)

Description of VSC-24

The Ventilated Storage Cask, designed by Sierra Nuclear Corporation, is a vertical concrete cask design that has been approved by the NRC and is in use at the Palisades nuclear power plant site. As with the NUHOMS design, this system relies on concrete for both neutron and gamma shielding and incorporates internal airflow passages requiring detailed shielding analyses to demonstrate acceptable streaming doses. The Ventilated Storage Cask design dose rates are 20 mrem/hr side contact and 50 mrem/hr top contact. A sketch of the Ventilated Storage Cask is shown in Figure F-7.

NRC Certification or Basis for License

The Sierra Nuclear Corporation's Model VSC-24 has been granted Certificate of Compliance Number 1004 by the NRC. The basis for this certificate is 10 CFR 72 Subparts L and K.

F.1.1.2.5 Manufacturers of Commercial Nuclear Fuel Dry Storage Systems Not Currently Licensed by the NRC in the United States

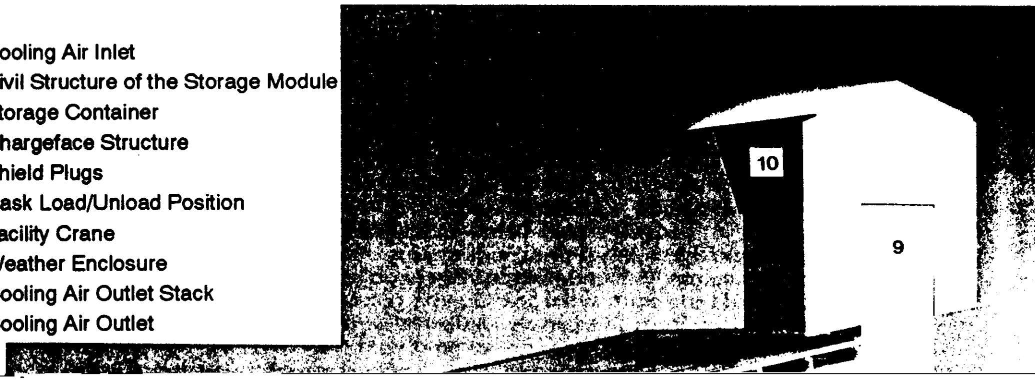
In addition to the above examples of dry cask storage systems licensed in the United States, there are other systems either licensed outside the United States or in the design and/or licensing stage (Table F-6). Tables F-7 and F-8 show shielding and thermal related parameters of the various dry cask models that are not currently licensed in the United States.

F.1.1.2.5.1 Description of MACSTOR

The MACSTOR system (designed by Atomic Energy of Canada, Ltd. and Transnuclear, Inc.), representing a synthesis of both metal and concrete casks in a modular dry vault, is being reviewed for use in Canada (AECLT, 1994). Spent nuclear fuel is placed in 0.95 cm (0.375 in) thick carbon steel canisters or baskets that are then placed (in a vertical position) in concrete modules. Air labyrinth passages into and

Key:

1. Cooling Air Inlet
2. Civil Structure of the Storage Module
3. Storage Container
4. Chargeface Structure
5. Shield Plugs
6. Cask Load/Unload Position
7. Facility Crane
8. Weather Enclosure
9. Cooling Air Outlet Stack
10. Cooling Air Outlet



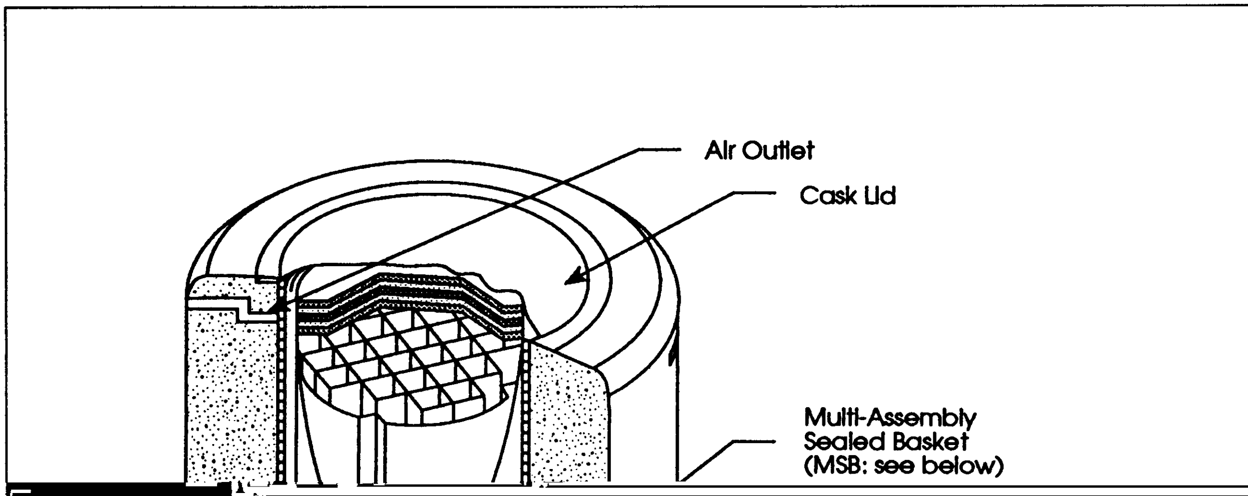

Air Outlet
Cask Lid
Multi-Assembly
Sealed Basket
(MSB: see below)

Table F-6 Manufacturers of Dry Storage Systems Not Currently Licensed in the United States

Manufacturer	*Facility or Model*	*Location or Status*
General Nuclear Systems, Inc. of Ontario Hydro	HDC	Canada
Atomic Energy of Canada, Ltd./Transnuclear Inc.	MACSTOR	Canada
Siemens Power Corporation	FUELSTOR	Germany
RISO National Laboratory	DR3	Denmark
Atomic Energy of Canada, Ltd.	SILO, Canister	Canada
Nuclear Assurance Corporation	NAC-26	10 CFR 72 License Pending
General Nuclear Systems, Inc.	X28 and X33	10 CFR 72 License Pending
Transnuclear, Inc.	TN-32	10 CFR 72 License Pending

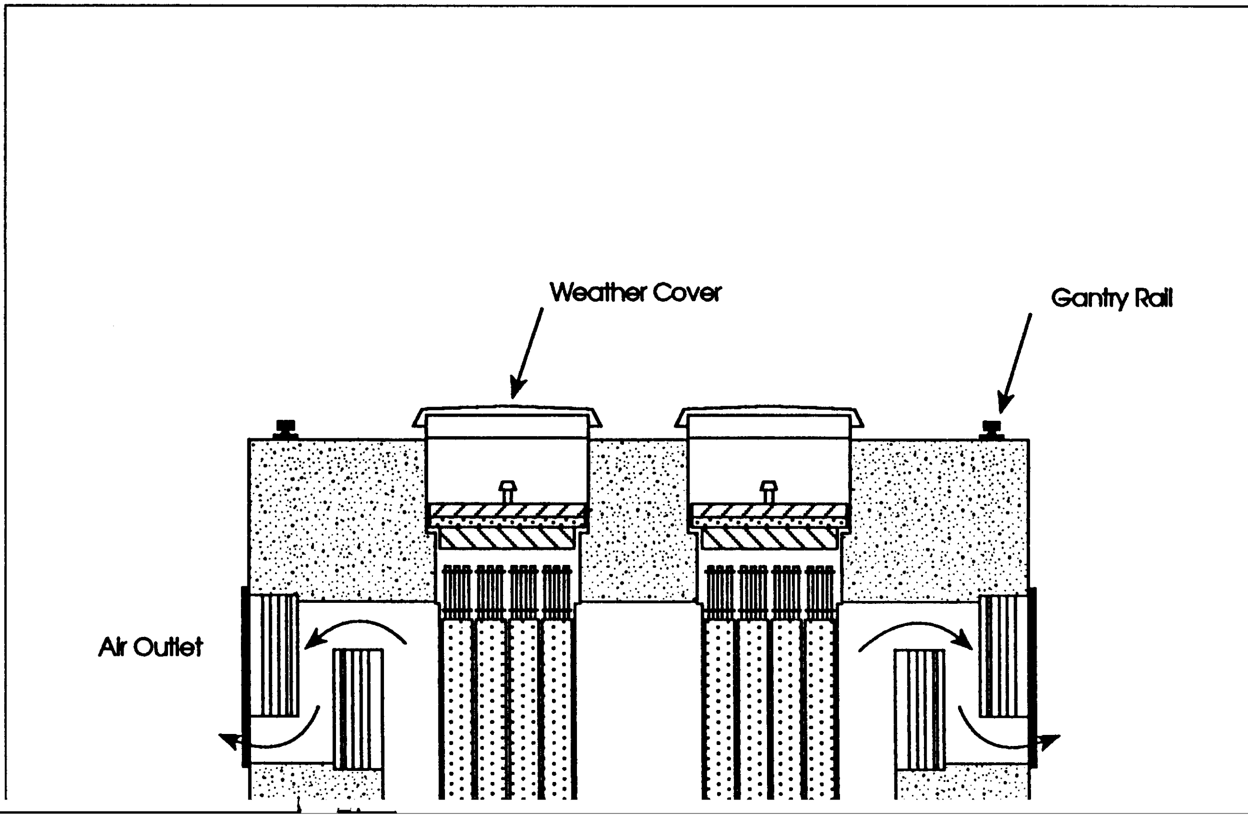
FUELSTOR = Fuel Encapsulation and Lag Storage

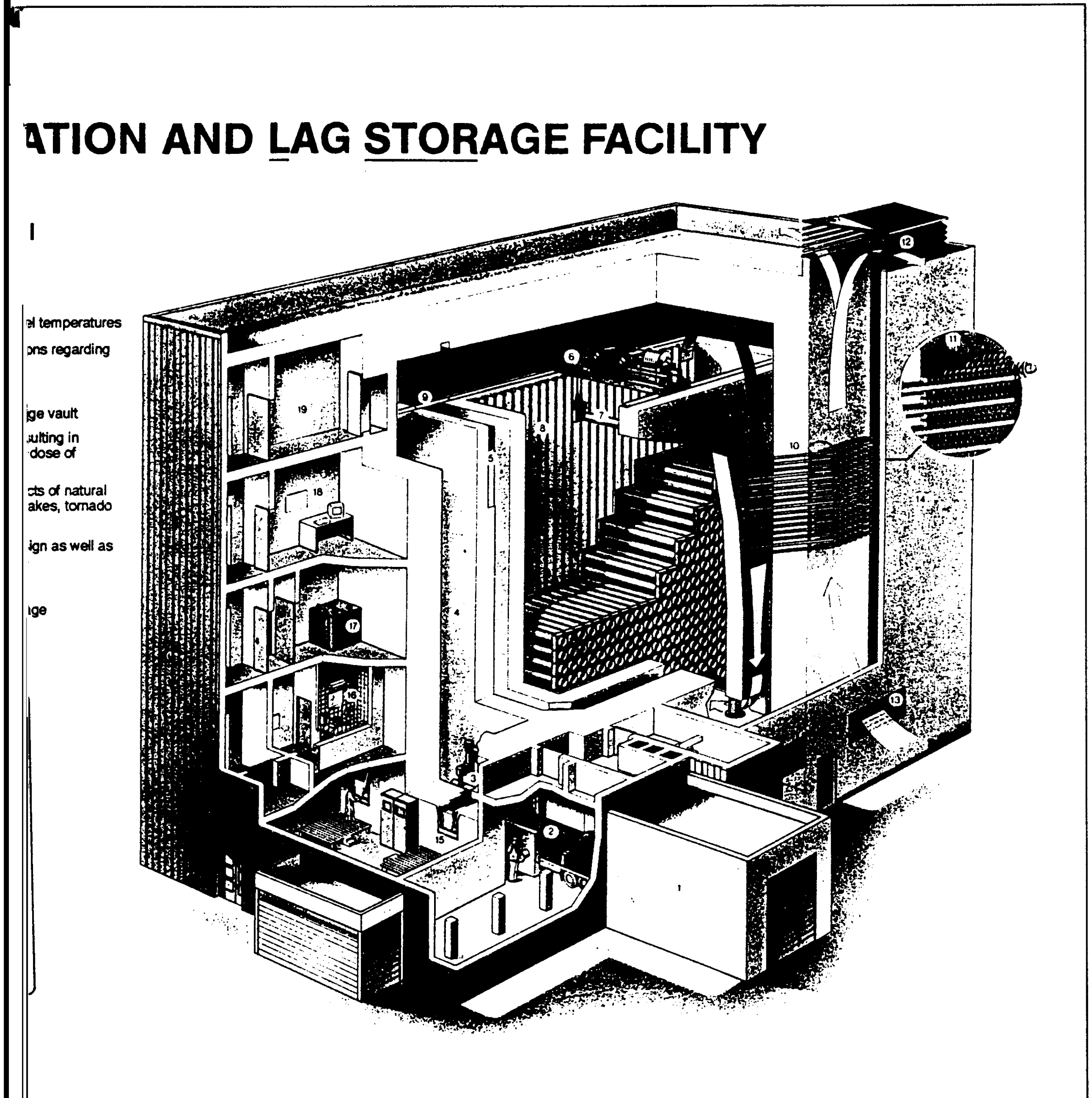
Table F-7 Comparison of Shield Design Parameters for Spent Nuclear Fuel Dry Storage Systems Not Currently Licensed in the United States

				Design Limit Surface Dose Rate

Table F-8 Comparison of Thermal Design Parameters for Spent Nuclear Fuel Dry Storage Systems Not Currently Licensed in the United States

Manufacturer	Model	Design Heat Load (KW)	Thermal Limits (°C)	Heat Transfer Mode(s)


Weather Cover
Gantry Rail
Air Outlet



Fuel Encapsulation and Lag Storage System Cross Section

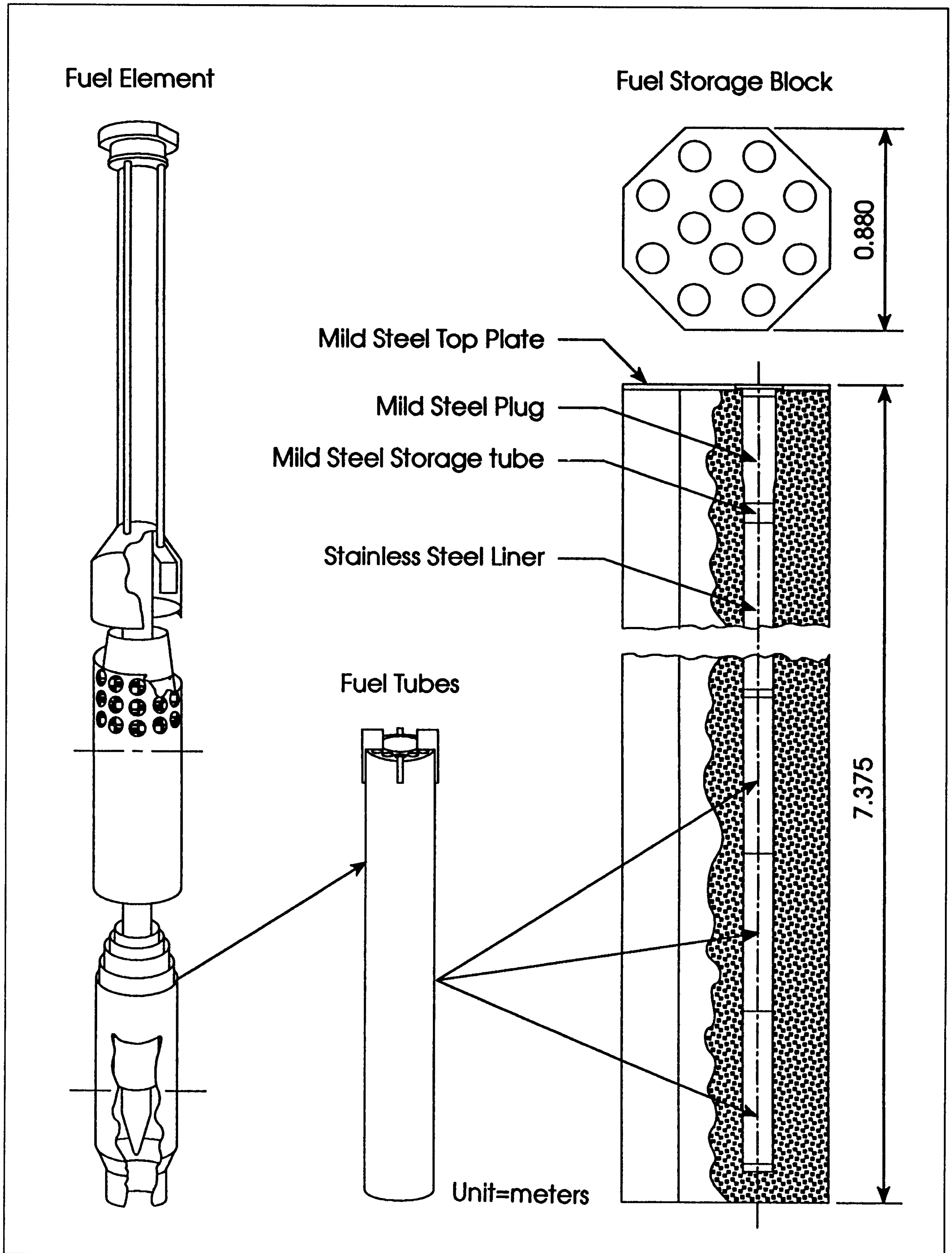


Figure F-10 RISO National Laboratory Design

research reactor spent nuclear fuel. The SILO has been licensed in Canada and is currently undergoing license approval in South Korea, which uses the same regulations as the NRC. A sketch of the SILO is given in Figure F-11.

F.1.1.2.5.5 Dual-Purpose Cask and Canister Systems

Dual-purpose designs must satisfy NRC requirements for both storage and transportation (10 CFR Parts 72 and 71, respectively). It is believed that such dual-purpose designs would reduce incident-free handling of individual spent nuclear fuel assemblies, reduce the volume of low-level radioactive waste that would otherwise be generated from using a single-purpose cask system (one cask for storage with subsequent transfer of individual assemblies to transport casks and disposal packages), and may play a role in reducing overall worker radiation exposures over a single-purpose cask system.

At the present time, there are two dual-purpose casks for light water reactor fuel use: Nuclear Assurance Corporation's Storage/Transport Cask and Vectra's dual-purpose canister system (MP-187). Nuclear Assurance Corporation's Storage/Transport Cask has received NRC approval. The NRC is expected to approve Vectra's MP-187 in the near future.

The VECTRA MP-187 is a derivative of a design approved earlier by the NRC. The MP-187 design includes a stainless steel confinement canister, a horizontal reinforced concrete module for storing the canister, and a special onsite/offsite transportation cask system that may also be used to store the canister in a vertical orientation. This system is currently being evaluated by the NRC for the Rancho Seco nuclear power plant. The applicant also has a variation for the canister design to accommodate canned spent nuclear fuel for damaged spent nuclear fuel assemblies, and which cannot be stored without a second confinement barrier.

DOE had proposed expanding the role of a dual-purpose system to that of a multi-purpose canister-based system (DOE, 1994f; DOE, 1994b; DOE, 1994c). Fuel would be loaded into a canister at the reactor site. The canister could then be placed into unique, specially designed overpacks for storage at the reactor site, transportation to a federal facility, or disposal in a repository. Final NRC approval for use of the multi-purpose canister as a component of the disposal package requires that the multi-purpose canister and its surrounding overpack meet 10 CFR Part 60 requirements. The fact that no site has been chosen yet for a repository adds an element of uncertainty to the third function: disposal. DOE has decided in November 1995 to withdraw its proposal to prepare the EIS for this canister. The Department of the Navy, however, will complete this EIS and will limit its scope to the storage and transport of Navy spent nuclear fuel.

F.1.2 Wet Storage Designs

In addition to the previous examples of dry storage technology, there are several types of wet storage systems currently in use at DOE sites and at commercial nuclear power facilities. These include aboveground pools (lined or unlined), inground pools (lined or unlined), and shutdown reactor vessels. For the purposes of this appendix, a pool refers to a canal or a basin.

Description of Wet Pool Spent Nuclear Fuel Storage Technology

The storage of spent nuclear fuel in pools (i.e., wet storage technology) has been in use for over 40 years (since the early water-cooled reactors began operating). The basic concept underlying wet storage is analogous to the development of light water-cooled nuclear reactors for defense, research, and electric power production purposes.

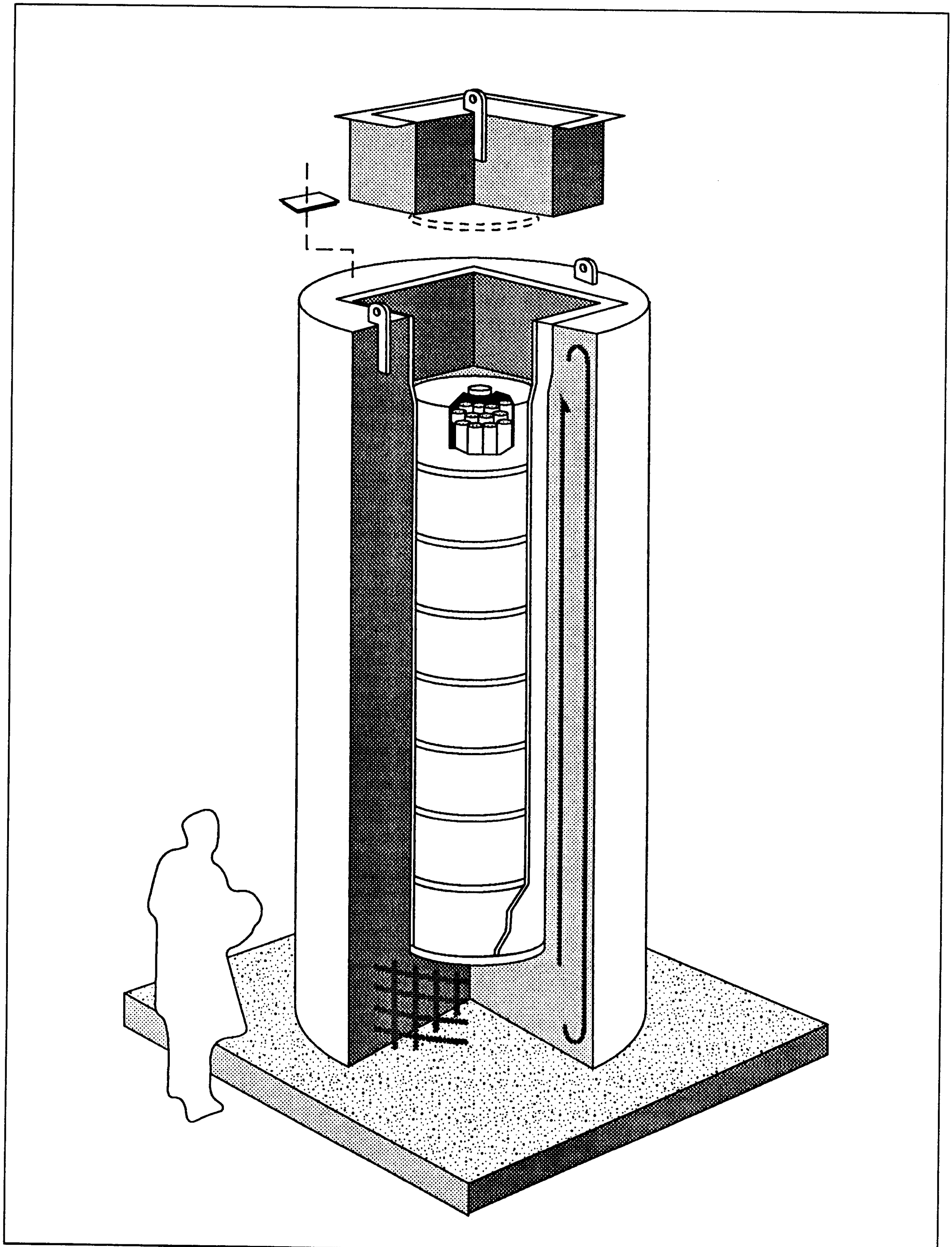


Figure F-11 The Atomic Energy of Canada, Ltd. Concrete SILO

In terms of spent nuclear fuel storage, water offers several distinct advantages, which can be summarized as:

- low cost for shielding and coolant medium,
- visual confirmation of fuel location and ease of handling,
- high heat capacity allowing for a large time period before thermal limits are exceeded,
- multi-purpose shield for both neutrons and gamma rays,
- inherent ability to retain many fission products which could leak from failed spent nuclear fuel, and
- insusceptibility to degradation from spent nuclear fuel radiation.

Water pool storage also has some shortcomings. These are:

- the need to maintain high purity water to prevent corrosion,
- the requirement for active safety systems connected to the water for heat removal, purity control, and water makeup,
- extensive lined and reinforced concrete walls for ensuring no leakage of water under all accident conditions,
- generation of radioactive waste from degraded fuel which is collected by the water purification systems, and
- groundwater monitoring to detect any leakage of radioactive pool water into the environment.

For every water-cooled reactor in the world, the decision has always been made to construct an adjoining or integral spent nuclear fuel storage pool. Currently, over 600 water cooled electric power-, research-, and defense-related reactors are operating in the world, each with its own wet storage pool for spent nuclear fuel (Nuclear Engineering International, 1993). Experience has shown that this technology is safe and effective.

At commercial nuclear power plants, the pool storage is located in a structure adjacent to a containment building that is capable of direct hydraulic connection to the reactor core through a system of canals, gates, and pools. The spent nuclear fuel pool building is designed and built to withstand all the accidents and dynamic loads required of other safety-related structures at nuclear power plants. It has its own crane and fuel handling equipment, and a separate heating, ventilation, and air conditioning system to mitigate radioactive releases to the environment. The nuclear power plant control room includes monitors and controls for the spent nuclear fuel pool. Redundant separate trains of equipment are used to fulfill the requirements of heat removal from the spent nuclear fuel pool water, removal of impurities and radioactive materials from the water, and maintenance of the water level to ensure adequate shielding above the spent nuclear fuel. At commercial power plants, such parameters as water level, water temperature, flow and temperature difference across heat exchangers used to cool the water, water purity, activity levels, and radiation dose rates are all monitored and measured.

All U.S. commercial nuclear power plant pools are stainless steel lined and use racks made of stainless steel to store spent nuclear fuel. Stainless steel is used to line the pool walls and floor to help maintain high water purity by preventing the release of chemicals from unlined concrete and to simplify decontamination at the end of the facility's life. The racks provide support and spacing for each fuel assembly, thus controlling criticality and maintaining fuel structural integrity.

Detailed criticality and thermal-hydraulic analyses are performed to demonstrate to the licensing authorities (the NRC in the United States) that fuel can never become critical, and that the assembly spacing in the racks allows for adequate cooling so as to prevent nucleate and bulk boiling in the pool or on any fuel surfaces. Shielding analyses substantiate the adequacy of the water depth above the fuel in the pool (usually at least 6.1 m or 20 ft), and the thickness of concrete pool walls and piping routing for systems connected to the pool water. This piping may contain pool water that is contaminated with radioisotopes released from spent nuclear fuel in the pool, and must be considered in dose rate evaluations. The shielding analyses provide assurances that the dose rate levels are acceptably low to workers around the spent nuclear fuel pool. Accident analyses are performed to show that the most conservative effects to the public of a postulated release of failed spent nuclear fuel fission products in the pool meet all regulatory dose rate limits.

One difference between nuclear power plant spent nuclear fuel wet storage and that which would be used for foreign research reactor spent nuclear fuel is that at nuclear plants, the pools include soluble boron in the water as a means of controlling criticality. Boron is a powerful neutron absorber, and as such prevents

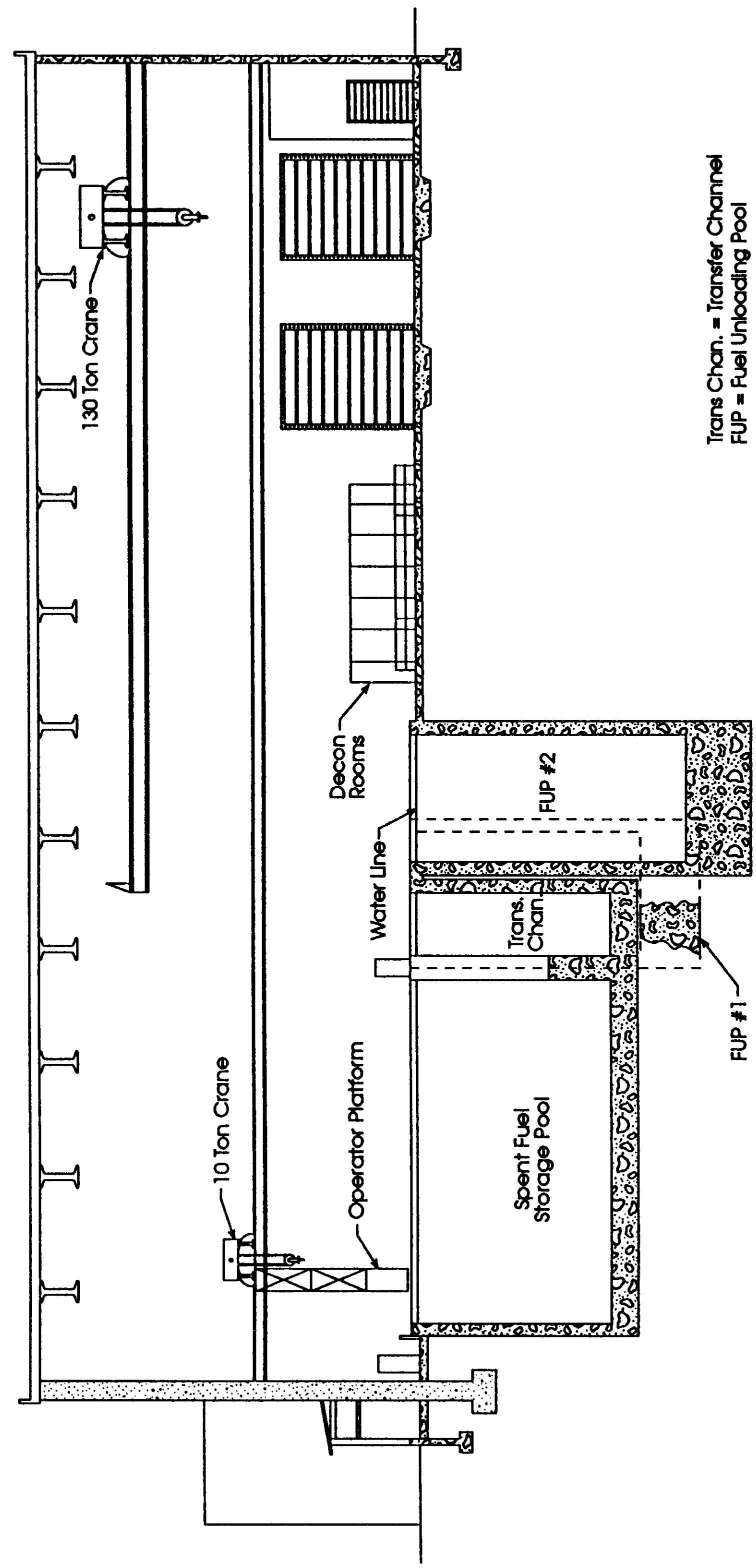


Figure F-12 Typical Wet Pool Storage Facility for Spent Nuclear Fuel

F.1.3 Summary of DOE Spent Nuclear Fuel Locations and Activities

DOE currently has about 2,700 MTHM of spent nuclear fuel in its storage facilities across the DOE complex (DOE, 1994h). Additional generation of about only 100 MTHM is anticipated during the next 40 years. Most of the spent nuclear fuel storage occurs at three sites: Hanford Site (77 percent), Idaho National Engineering Laboratory (10.9 percent), and Savannah River Site (7.3 percent) (Table F-9). Note that the quantities of DOE spent nuclear fuel completely dwarf the expected amount of foreign research reactor spent nuclear fuel (about 19 MTHM) on an MTHM basis (i.e., foreign research reactor spent nuclear fuel is less than 1 percent of the total). However, on a volume basis, foreign research reactor spent nuclear fuel represents about 10 percent of the total and, thus, their storage facilities would be of a significant size. Predominantly wet storage is used at DOE sites, although some limited experience exists with dry storage (e.g., Los Alamos National Laboratory and Idaho National Engineering Laboratory).

Table F-9 DOE Spent Nuclear Fuel Inventory[a, b]

Generator or Storage Site[c]	Existing (1995)		Future Increases (through 2035)		Total (2035)	
	MTHM[d]	Percent	MTHM[d]	Percent	MTHM[d]	Percent
Hanford Site	2,132.44	80.6	0.00	0.0	2,132.44	77.8
Idaho National Engineering Laboratory[e]	261.23	9.9	12.92	13.5	274.14	10.0
Savannah River Site	206.27	7.8	0.00	0.0	206.27	7.5
Naval Nuclear Propulsion Reactors	0.00[f]	0.0	55.00	57.6	55.0	2.0
Oak Ridge Reservation	0.65	<0.1	1.13	1.2	1.78	<0.1
Other DOE Sites	0.78	<0.1	1.50	1.6	2.28	<0.1
Non-DOE Domestic Research Reactors[g]	2.22	<0.1	3.28	3.4	5.50	0.2
Special-Case Commercial Reactors[h]	42.69	1.6	0	0	42.69	1.6
Foreign Research Reactors[i]	0	0	21.7	22.7	21.70	0.8
Total	**2,646.27**		**95.53**		**2,741.80**	
Percent of 2035 Total	**96.5**		**3.5**		**100.00**	

[a] *Source: DOE, 1995g*

[b] *Numbers may not sum due to rounding.*

[c] *The Nevada Test Site does not currently store or generate spent nuclear fuel and is not expected to generate spent nuclear fuel through 2035. However, in the 2010-2020 timeframe, a repository may open, with annual capacity over 1,000 MTHM.*

[d] *One MTHM equals approximately 2,200 pounds.*

[e] *Sum of fuel located at the Idaho National Engineering Laboratory.*

[f] *Existing inventory of Naval spent nuclear fuel is included in the Idaho National Engineering Laboratory totals (9.95 MTHM).*

[g] *Includes research reactors at commercial, university, and Government facilities.*

[h] *This total is just that stored at non-DOE facilities (Babcock & Wilcox Research Center and Fort St. Vrain). The total inventory of spent nuclear fuel from special-case commercial reactors is 186.41 MTHM. This fuel is also stored at the Idaho National Engineering Laboratory, the Oak Ridge Reservation, the Hanford Site, the Savannah River Site, and the West Valley Demonstration Project.*

[i] *At the Savannah River Site and the Idaho National Engineering Laboratory.*

Wet Storage

DOE spent nuclear fuel pools are in many cases more than 20 years old and were originally unlined, due to simplicity and the relatively short planned duration (3 to 6 months) of spent nuclear fuel storage prior to reprocessing. The spent nuclear fuel storage basins are concrete with 30 to 90 cm (1 to 3 ft) thick walls, and the bottom is usually thicker than the sides. For shielding purposes, the pool maintains a minimum of 3 m (10 ft) of water over the spent nuclear fuel at all times. Thus, total water depth typically ranges between 4.5 to 6.1 m (15 to 20 ft), although some facilities extend to 9.1 m (30 ft). Steel, stainless steel, or aluminum racks are affixed to the bottom of the pool for holding the spent nuclear fuel in a vertical configuration. The spent nuclear fuel basins provide for recirculation and heat removal capabilities, but limited water clarification and purification. Chemicals from the exposed concrete increase pool turbidity

and tend to accelerate corrosion phenomena, particularly for aluminum-clad fuels. Some of the DOE spent nuclear fuel wet storage facilities do not meet the present construction requirements.

Wet storage still remains the predominant technology for storing irradiated materials (DOE, 1993b; Taylor et al., 1994). Currently, there are some 29 DOE spent nuclear fuel storage pool facilities in the complex, ranging in age from 10 years to more than 40 years. Facilities built more than 30 years ago were constructed to standards far less rigorous than exist today. Several DOE orders address spent nuclear fuel storage facilities indirectly, while DOE Order 6430.1A specifically sets the design criteria that addresses storage facilities (spent nuclear fuel facilities that are part of a reactor facility are covered by DOE Order 5480.6). Most DOE storage pools were not designed for long-term storage of spent nuclear fuel and targets and have very limited space available for consolidation.

Most of the storage pool surfaces are bare concrete A few are lined with stainless steel, and some are coated with epoxy or vinyl. The unlined pools are more susceptible to leakage and to increased contamination by soluble radionuclides. The unlined bare concrete storage pools do not have effective leak-detection systems to detect and capture potential leaks. To help identify pool leakage, more than 50 percent of DOE storage pools have had groundwater monitoring wells installed.

Severe corrosion of materials within many of the DOE storage pools has occurred. Corrosion has been generally attributed to poor water quality control and material incompatibilities, which has led to pitting and galvanic corrosion of spent nuclear fuel and storage equipment. This could potentially create a problem when the spent nuclear fuel materials have to be moved. In some cases, equipment failure could cause fissile material reconfiguration, which could increase nuclear criticality concerns. As a result of corrosion, release of radionuclides and fissile material to the pools has occurred. Corrosion also creates handling, packaging, inventory control, waste generation, and cleanup problems with the storage pools.

Savannah River Site and DOE (Taylor et al., 1994) provide the following facilities potentially suitable f

- 105-L Disassembly Basin
- 105-C Disassembly Basin
- 105-P Disassembly Basin
- Receiving Basin for Offsite Fuels (RBOF) Facility (244-H)
- BNFP (acquisition required).

However, only the ICPP-666 pool and the BNFP were found to meet all current standards, and, thus, be considered suitable for long-term storage.

DOE has improved some of its spent nuclear fuel facilities and has plans for additional upgrades (DOE, 1993b; DOE, 1995g). Typical upgrades include:

- installation and operation of water purification equipment, such as demineralizer columns and filters,
- reracking and fuel consolidation to increase fuel storage space, and
- improving seismic resistance (where possible, via additional supports).

These upgrades would extend the life of existing facilities and allow safe storage of spent nuclear fuel until new facilities are constructed or the spent nuclear fuel is chemically separated. In addition, spent nuclear fuel suspect of leaking during this interim period would be removed and canned to extend its safe storage.

Dry Storage

DOE has fewer dry storage facilities, and these range from approximately 1 to 50 years in age. There are many different types and applications of dry storage used throughout the DOE complex. Spent nuclear fuel is sorted in steel structures; lined and unlined concrete hot cells; steel-lined; concrete; below-grade vaults; reprocessing canyon dissolver cells; cans contained in steel wells; and large, above-grade storage casks. Spent nuclear fuel has been characterized and stored in dry configurations within hot cell facilities since the 1950s. Most DOE hot cells were not designed and built for long-term storage of spent nuclear fuel. Their primary mission was to conduct tests and basic research on irradiated fuels resulting in very limited capacity for storage of spent nuclear fuel.

Since the 1970s, spent nuclear fuel has been stored in facilities specifically engineered for longer-term dry storage. Modern dry storage methods in newer facilities provide low corrosion environments within sealed barriers for monitored interim retrievable storage. A few examples of dry storage confinement methods include sealed canisters in wells surrounded by concrete and extensive release protection incorporating High Efficiency Particulate Air-filtered ventilation systems. By using current dry storage technology, dry storage facilities could be engineered to withstand severe natural phenomena hazards, fires, and explosions without damage to the fuel or release of radionuclides. Dry storage technologies can be adapted to store many types of damaged and undamaged DOE-owned spent nuclear fuel.

The application of dry storage technologies generally results in fewer environmental, safety, and health issues as compared with wet storage. However, DOE has limited experience with aluminum-clad, high decay heat fuels in dry storage facilities.

Some quantities of spent nuclear fuel may be in dry storage facilities for much longer than originally anticipated. Over the years, several inground steel-lined storage well barriers have had the potential for severe corrosion, which could result in undetected releases to the environment. This is particularly important, because of the inaccessibility of these facilities for inspection and characterization (e.g., Argonne National Laboratory Radioactive Scrap and Waste Facility).

The Savannah River Site and DOE (Taylor et al., 1994) consider the following facilities suitable for near-term dry storage of spent nuclear fuel.[1]

- Argonne National Laboratory-West
  - Hot Fuel Examination Facility
  - Radioactive Scrap and Waste Facility
- Idaho National Engineering Laboratory
  - Test Area North Test Pad
  - ICPP Irradiated Fuel Storage Facility (IFSF)
  - ICPP-749 (Drywells, Second Generation)
- Savannah River Site
  - 221-H (Extensive modification required)
  - 221-F (Extensive modification required).

However, certain DOE requirements, such as DOE Orders 6430.1A and 5480.6, make it likely that these facilities could not qualify for future, long-term, dry storage of spent nuclear fuel. Excluding the Savannah River Site facilities (because of the extensive required modifications), none of these facilities appear to be very useful for long-term spent nuclear fuel storage. Extensive modifications to the facilities would be required to meet seismic criteria and increase the storage capacity or convert existing facilities (e.g., F- and H-Canyons at the Savannah River Site) into suitable dry storage facilities. However, facilities such as the Hot Fuel Examination Facility and Test Area North appear suitable as possible staging and characterization facilities in a dry cask storage approach, based upon the presented information (Taylor et al., 1994).

F.1.3.1 Savannah River Site

The Savannah River Site occupies an area of approximately 800 km^2 (310 mi^2) in South Carolina, in a generally rural area about 40 km (25 mi) southeast of Augusta, Georgia (DOE, 1995g). The Savannah River forms the southwestern border of the Savannah River Site. The Savannah River Site consists primarily of managed upland forest with some wetland areas, and facilities and railways occupy approximately five percent of the Savannah River Site land area. Figure F-13 presents a map of the Savannah River Site with spent nuclear fuel facilities displayed.

1 *Existing facilities in Nevada were not included in the analysis.*

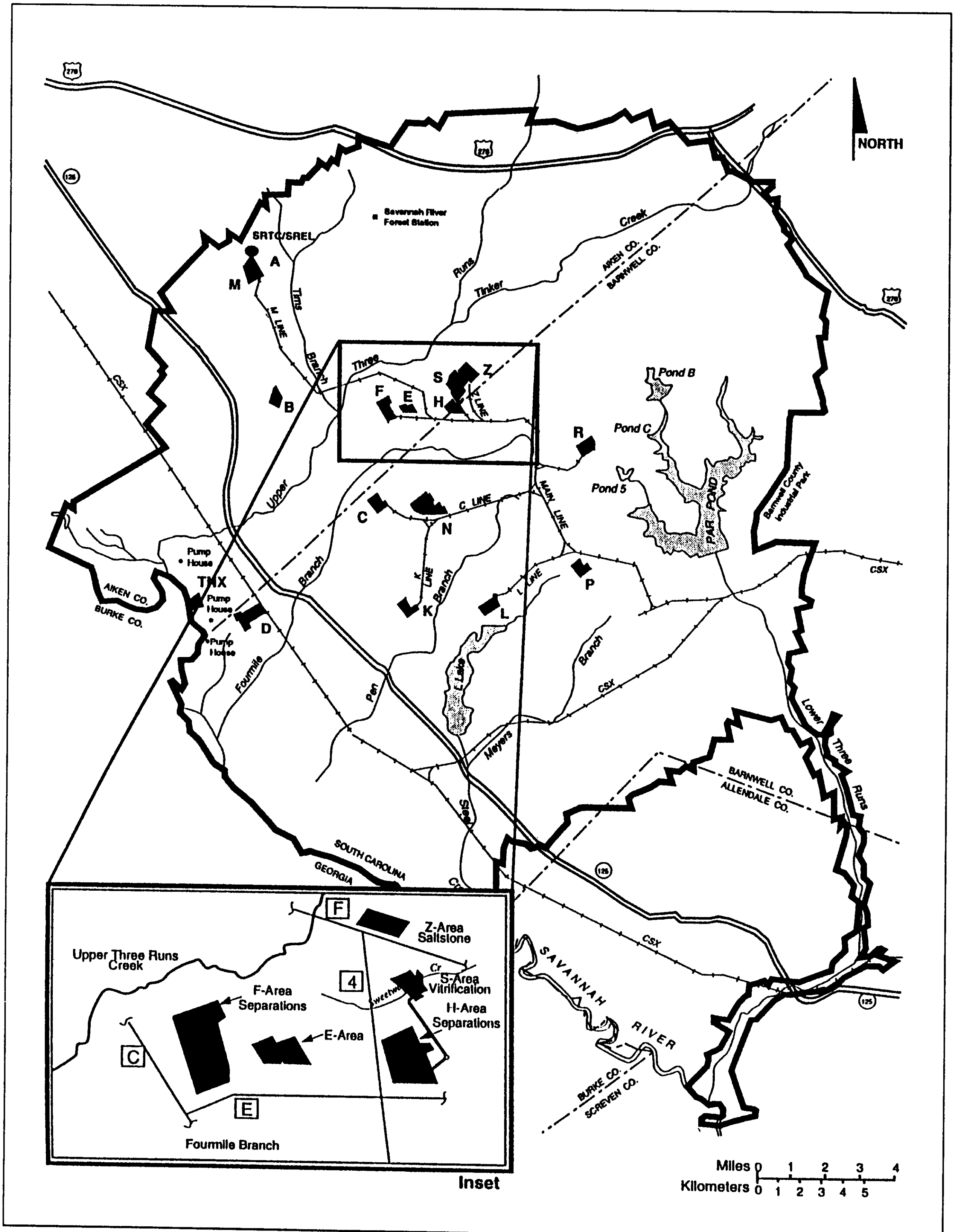


Figure F-13 Location of Principal Savannah River Site Facilities

The primary Savannah River Site facilities were used for the production of nuclear materials. Currently, the production reactor facilities are not operating and are in either shutdown or standby mode. Several large waste management projects are now underway at the site, including the Defense Waste Processing Facility for the vitrification of high-level waste.

F.1.3.1.1 Spent Nuclear Fuel Activities at the Savannah River Site

The Savannah River Site currently stores approximately 201 MTHM of spent nuclear fuel (DOE, 1995g), or approximately 7 percent of the DOE total, including the following:

- 184.4 MTHM of aluminum-based spent nuclear fuel, including plutonium target material

- 4.6 MTHM of commercial spent nuclear fuel (zircaloy-clad),

- 11.9 MTHM of test and experimental reactor, zircaloy-clad fuel, and

- 5.4 MTHM of test and experimental reactor, stainless steel-clad fuel.

This fuel is stored in several basins onsite. The F- and H-Area Canyons are the processing and separations facilities at the Savannah River Site, and each has a small associated wet storage basin. Three reactor disassembly basins (K, L, and P) contain the reactor fuel and target materials. A fourth reactor disassembly basin (C) currently is the only basin without security upgrades necessary for any storage activities. These basins consist of unlined concrete with inadequate water purification equipment for extended storage of aluminum-clad spent nuclear fuels. These reactor basins were built in the 1950s and were not intended for the long-term storage ("years") of radioactive materials. Furthermore, poor water chemistry has corroded some of the spent nuclear fuel in the K- and L-Reactor disassembly basins, resulting in the release of fissile materials to the pool water. Also, these reactor basins are not seismically qualified and lack modern earthquake resistant features. Ongoing facility upgrades of the L-Reactor disassembly basin are intended to correct the conditions of the basin. Deionization of the basin has lowered the conductivity to acceptable levels for corrosion control. Lower conductivity would greatly reduce the probability of new corrosion and reduce the rate of progression of existing corrosion. The control of the conductivity after the completion of the deionization would be accomplished using the Disassembly Basin Upgrade Project which was initiated to address near term activities and vulnerabilities associated with storing fuel in the L-Reactor disassembly basin. With the upgrades to be completed by mid-1996 (Miller et al., 1995), the L-Reactor basin can be expected to safely store spent nuclear fuel for as long as 10 to 20 years. These upgrades include the following:

- A continuous on-line deionization system to improve water chemistry. The continuous deionization system will lower and control the conductivity levels of the basin thereby minimizing corrosion. The continuous deionizer system also removes ionic radionuclide concentrations, specifically Cesium-137.

- A makeup water deionizer to improve the quality of makeup water supplied to the basin. This action will mitigate any additional load on the continuous deionization system.

- New equipment and systems for alternative packaging and removal of waste.

A Basis for Interim Operation document for the L-Reactor in cold standby conditions was prepared by the Westinghouse Savannah River Company (WSRC, 1995b). The Basis for Interim Operation addressed the effects of process events on the facility worker and the effects of process and natural phenomena hazards events on the public and the environment. The Basis for Interim Operation document concluded that the facility could continue to operate within the safety envelope, identified in the Basis for Interim Operation, without undue risk to the public or the environment.

The RBOF is the other major facility for spent nuclear fuel storage. The RBOF is more suitable than the reactor basins because it is lined (epoxy sides, stainless steel bottom) and has a water purification system. The BNFP, after refurbishing, would be suitable for foreign research reactor spent nuclear fuel storage because it is fully lined with stainless steel, has water purification systems, and has active heat removal systems. Major spent nuclear fuel storage facilities are summarized in Table F-10.

Table F-10 Major Savannah River Site Spent Nuclear Fuel Storage Facilities

Facility	*Characteristics*	*Capacity for Foreign Research Reactor Spent Nuclear Fuel Elements*	*Access*
105-K Disassembly Basin	Basin Dimensions: 46.9 x 65.8 x 5.2m (154'W x 216'L x ~17'D) Basin Water: 13.2 million l (3.5 million gal)	None initially 20,000 after upgrades	Truck/Rail
105-L Disassembly Basin	Basin Dimensions: 46.9 x 65.8 x 5.2m (154'W x 216'L x ~17'D) Basin Water: 13.2 million l (3.5 million gal)	None initially 20,000 after upgrades	Truck/Rail
105-C Disassembly Basin	Basin Dimensions: 39.6 x 58.2 x 5.2m (130'W x 191'L x ~17'D) Basin Water: 13.6 million l (3.6 million gal)	None initially 20,000 after upgrades	Truck/Rail
105-P Disassembly Basin	Basin Dimensions: 55.5 x 68.2 x 5.2m (182'W x 223'L x ~17'D) Basin Water: 18.2 million l (4.8 million gal)	None initially 20,000 after upgrades	Truck/Rail
RBOF (244-H)	Basin 1: 8.2 x 12.1 x 6.7m depth over two-thirds of floor space 8.8m depth over one-third of area Basin 2: 8.2 x 3.9 x 8.8m depth Basin Water: 1.7 million l (450,000 gal)	~1000 initially, plus 1,425 after rearranging[b]	Truck/Rail
BNFP[a]	Several Pools: Main Pool: 14.6 x 14.6.x 9.8m (48'L x 48' x 32'D) Basin Water: 2.1 million l (550,000 gal)	None initially 25,000 after acquisition and reactivation[b]	Truck/Rail

[a] *Discussed in more detail in Section F.1.3.1.3; rail spur not currently active but would be included in reactivation.*

[b] *Difference in capacity between RBOF and BNFP is due to greater pool depth of BNFP and different fuel packing density assumptions for the two facilities.*

F.1.3.1.2 Spent Nuclear Fuel Storage Facilities Available for Foreign Research Reactor Spent Nuclear Fuel at the Savannah River Site

The RBOF is the principal facility applicable for foreign research reactor spent nuclear fuel. This basin has been operating and receiving spent nuclear fuel, including foreign research reactor spent nuclear fuel, since 1964, and is located in H-Area, near the center of the Savannah River Site. The 1,393 m^2 (15,000 ft^2) facility consists of an unloading basin, two storage basins, a repackaging basin, a disassembly basin, and an inspection basin. The basins and their interconnecting canals hold approximately 1,893,000 l (500,000 gal) of water. Spent nuclear fuel elements arrive in lead-lined casks weighing from 22 to 64 metric tons (24 to 70 tons), which a crane lifts from a railroad car or a truck trailer and places in the unloading basin. About 30 percent of the fuels in the RBOF consist of uranium clad in stainless steel or zircaloy, which the Savannah River Site facilities cannot process without modifications. The RBOF is discussed in more detail in Section F.3.

In March 1995, the Savannah River Site estimated that the RBOF has the capacity for approximately an additional 1,000 spent nuclear fuel elements (O'Rear, 1995). However, the Savannah River Site has determined that 1,425 additional spaces can be made available by rearranging fuel in the pools and moving spent nuclear fuel to other storage areas, such as one of the reactor disassembly basins. If empty, the total RBOF capacity would be 6,500 foreign research reactor spent nuclear fuel elements.

F.1.3.1.3 Planned or Potential Spent Nuclear Fuel Storage Facilities at the Savannah River Site for Foreign Research Reactor Spent Nuclear Fuel

The Savannah River Site is evaluating the use of several new planned or potential facilities for foreign research reactor spent nuclear fuel management. These include:

- a modular dry vault storage building,
- dry cask storage, or
- wet pool storage.

These technologies may require additional support facilities for such functions as: spent nuclear fuel examination, spent nuclear fuel characterization, cask loading and unloading, spent nuclear fuel repackaging, and cask maintenance. The Savannah River Site is also evaluating the use of one or more of the reactor disassembly basins for near-term wet storage of foreign research reactor spent nuclear fuel. These facilities are discussed in more detail in Section F.3.

The Savannah River Site is also evaluating the potential storage of spent nuclear fuel at the BNFP facility. Allied General Nuclear Services constructed a large reprocessing facility for commercial spent nuclear fuel in Barnwell, South Carolina, adjacent to the Savannah River Site (Fields, 1994; Matthews, 1994 and 1991; Taylor et al., 1994; Williams, 1994; WSRC, 1992a-d). This plant was never operated due to a change in Government policy, and was mothballed in the 1980's. The BNFP includes a wet fuel storage basin that is approximately twice the area and potentially has over four times the spent nuclear fuel capacity of the RBOF facility at the Savannah River Site. The wet storage basin is fully lined and seismically qualified and would be capable of storing all of the currently identified foreign research reactor spent nuclear fuel (Jackson, 1994). Facility acquisition, replacement of removed equipment, reactivation

installation of suitable storage racks, and checkout at the facility would be required prior to its use.

Figure F-14 displays the location of the BNFP in relation to the Savannah River Site. This land was originally part of the Savannah River Site. The BNFP site consists of approximately 680 hectares (ha) (1,680 acres).

Allied General Nuclear Services designates the fuel pool area of the plant as the "Fuel Receiving and Storage Stations." Considerable documentation exists for the facility, including the engineering designs, the Environmental Impact Statement (EIS), and the Final Safety Analysis Report submitted to the NRC. The pools and attendant cranes are fully seismically qualified structures. The pool section includes ion exchange systems for pool water purification and a separate radwaste system (solidification may need to be added). The section incorporates capabilities for receipt of either truck or railcarried casks. The main crane is rated at 122 metric tons (135 tons).

The Fuel Receiving and Storage Station facility is shown in Figure F-15 and was designed and constructed to receive, store, and handle spent (irradiated) light water reactor fuel. Spent nuclear fuel assemblies are received in shielded casks by either truck or rail. The assemblies are unloaded underwater and stored underwater to provide cooling and shielding. Stored fuel can be remotely transferred to the adjacent

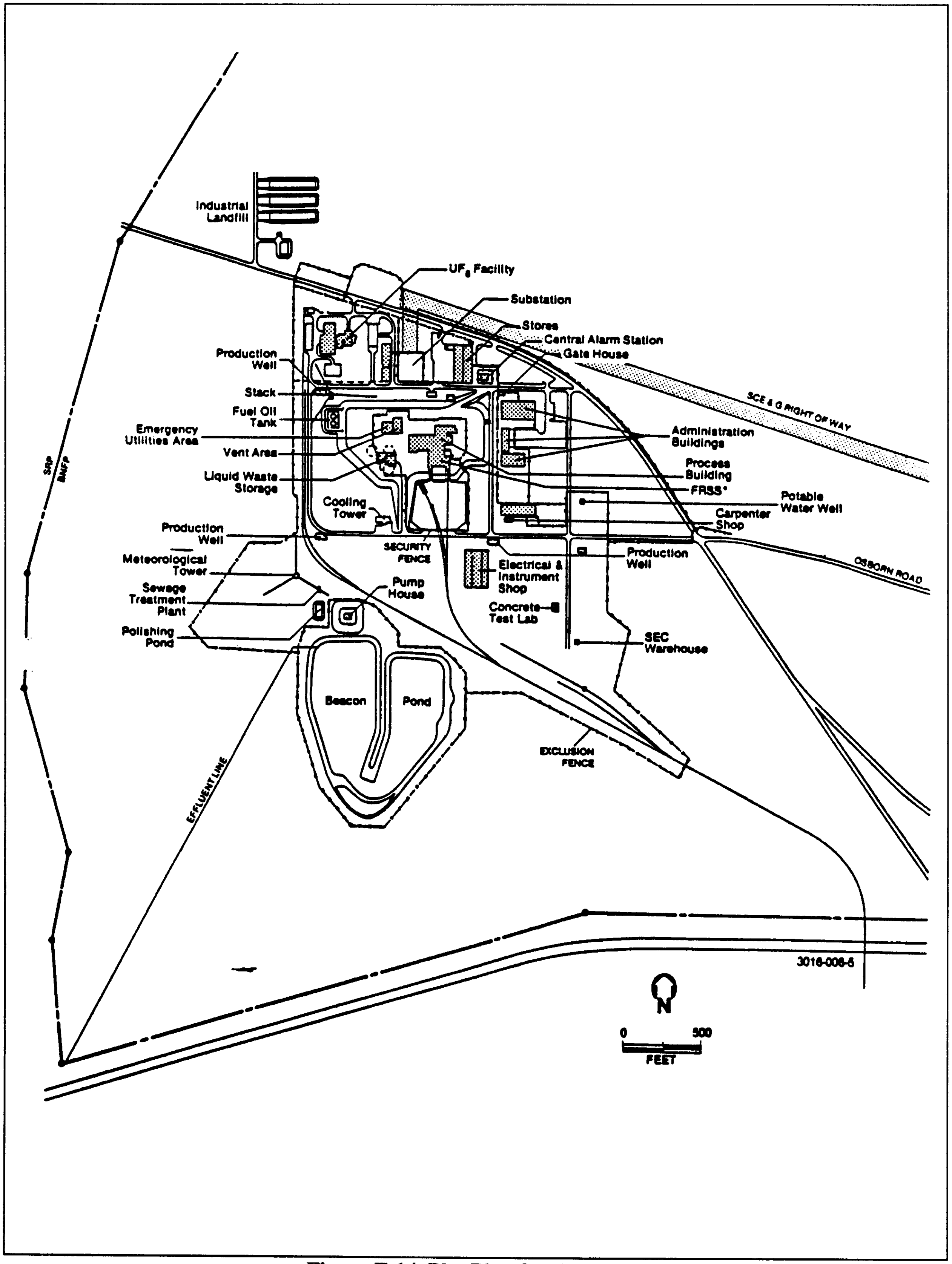


Figure F-14 Plot Plan for the BNFP

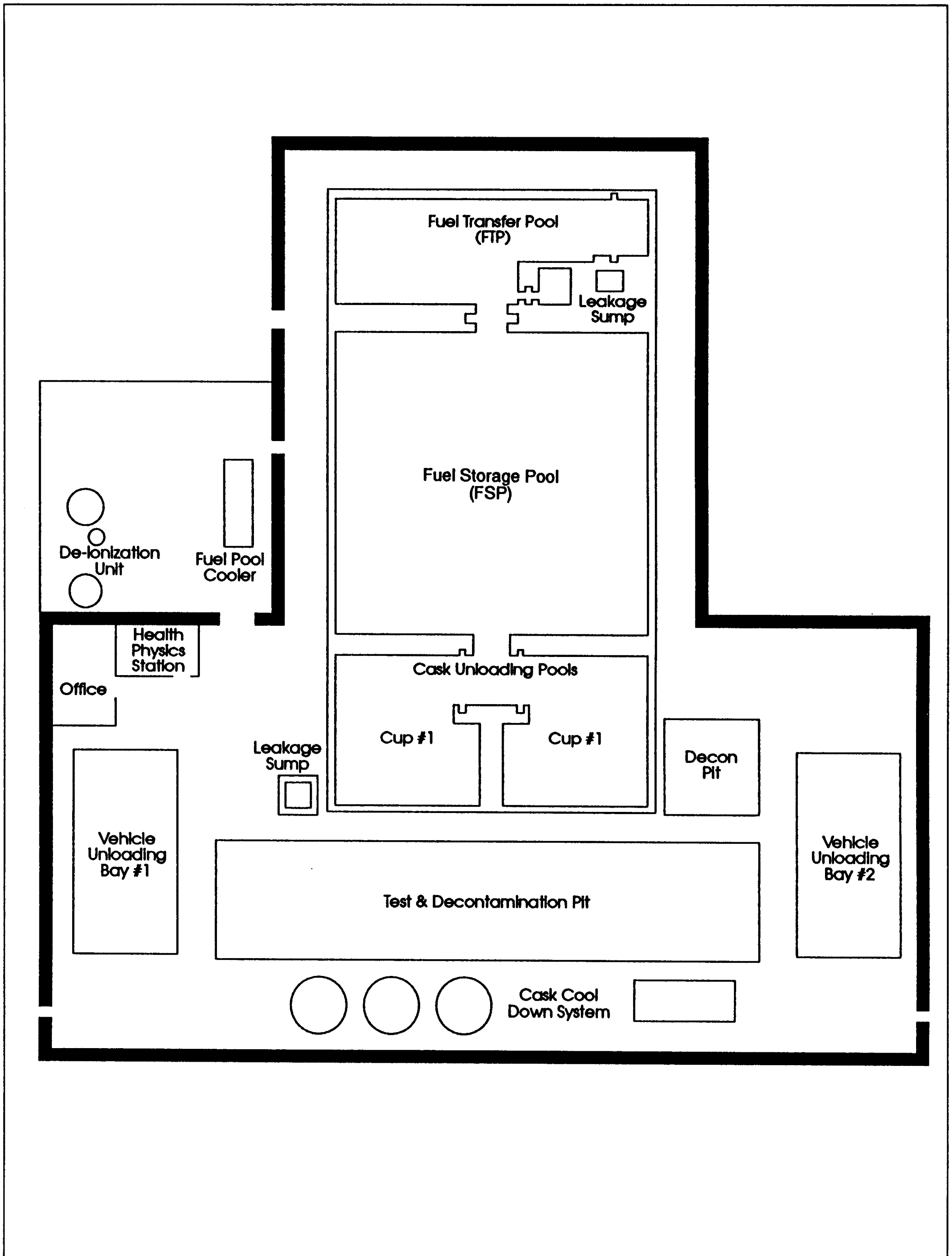


Figure F-15 Schematic of a Fuel Receiving and Storage Station at BNFP

The following areas of the Fuel Receiving and Storage Station are safety class structures:

- pool concrete structure,
- pool and crane column foundations,
- embedments for the fuel storage racks,
- crane rails, rail supports, and restrainer bars which retain the cranes on their rails and prevent their falling into the pools,
- cask barrier beams and embedments,
- energy absorbing pads in the Cask Unloading Pools,
- emergency water supply line, and
- Fuel Receiving and Storage Station walls to the 7.6 m (25 ft) level above grade. Clean spent nuclear fuel casks are moved to the Fuel Receiving and Storage Station water pool area. This area is divided into six pools consisting of two Cask Unloading Pools, one Fuel Storage Pool, one Failed Fuel Pool, one Fuel Transfer Pool, and an examination cell/pool.

Water shielding of 3.7 m (12 ft) is provided in the Fuel Receiving and Storage Station pools. This limits surface dose rates to a calculated 0.08 mrem/hr, assuming design basis Light Water Reactor fuel, and permitting at least 40 hours per week working time for an operator. Handling systems are designed with special limit switches and mechanical stops to prevent raising fuel higher than the design depth of the shielding water.

The water in the five pools of the Fuel Receiving and Storage Station is channeled and treated to promote maximum clarity, to control temperature, and to minimize corrosion and radioactivity. This is accomplished by continuous filtration through 95 percent efficient 5 micron pore size filter elements, cooling in heat exchangers to hold the pool water temperature below 41°C (105°F), and demineralization.

Demineralizing water treatment is designed to maintain radioactivity levels below 0.0005 μCi/ml. Pool water is pumped from the Fuel Storage Pool at 7,570 l/min (2,000 gal/min), directed through the heat exchangers, and returned to the Fuel Storage Pool. A second stream is pumped at 1,135 l/min (300 gal/min) from a pool and is filtered. After filtration, one-half of this stream is treated by ion exchange. The combined filtered and purified solution is then returned to the Fuel Storage Pool. The pool piping system is arranged so that the cleanup stream can be removed from or returned to any of the pool areas, permitting cleanup of contaminated water.

The cooling system is designed to remove heat at a rate of 4,000 kilowatts (14 million Btu/hr). The cooling capacity can be increased by expanding the capacity of the heat exchanger system in the Fuel Receiving and Storage Station. The estimated life for the structure is 50 years (Fields, 1994; Matthews, 1994; Taylor et al., 1994).

The Fuel Receiving and Storage Station has the following six pools:

- two cask pools, each 18.3 m (60 ft) deep,
- failed fuel pool (for degraded fuel),
- fuel transfer pool, 18.3 m (60 ft) deep,

- examination cell/pool, and
- main pool, 14.6 m x 14.6 m x 9.8 m deep (48 ft by 48 ft by 32 ft deep).

All of the pools are lined with stainless steel and are designed to maintain a minimum of 3.7 m (12 ft) of water above the fuel for shielding. The pools include detectors and flow channels for managing potential leaks. The original capacity of the main pool was 400 MTHM. Various analyses have been performed to increase this capacity to the 1,200 to 2,000 MTHM range with reracking and other arrangements. It has been estimated that maximum wet storage corresponds to approximately 5,200 Pressurized Water Reactor assemblies (Taylor et al., 1994). For foreign research reactor spent nuclear fuel, this would correspond to over 25,000 elements; and, thus, as noted previously, the BNFP could accommodate all of the fuel.

The environmental impacts of spent nuclear fuel storage at the BNFP have also been analyzed for between 360 and 5,000 MTHM of commercial fuel (Taylor et al., 1994). The results were:

- Dose commitments to the 80 km (50 mi) population were estimated to be 0.067 person-rem and 0.071 person-rem for 15- and 25-year storage periods, respectively.
- The worst accident would result in a dose commitment of 1 mrem total body, 6 mrem thyroid, and 100 mrem skin to an exposed individual located at the eastern boundary of the site.

These analyses were based upon commercial spent nuclear fuel, but should bound the consequences of foreign research reactor spent nuclear fuel storage at the BNFP. Potential impacts are discussed in more detail in Section F.4.

The BNFP site consists of some 680 ha (1,680 acres), bounded on three sides by the Savannah River Site. Preliminary walkthroughs and analyses by the Savannah River Site indicate the facility is in good condition, and principally needs a main transformer for power supply. The Savannah River Site has estimated a cost of $50 million (Matthews, 1991; WSRC, 1992a-d). Actual acquisition and reactivation costs are claimed to be as low as $25 million (Matthews, 1994; WSRC, 1992a-d). This facility, however, would not be available immediately to receive the foreign research reactor spent nuclear fuel.

Figure F-16 displays the foreign research reactor spent nuclear fuel storage capacity versus time for the Savannah River Site. Clearly, the Savannah River Site can accommodate foreign research reactor spent nuclear fuel at existing facilities supplemented by dry storage, modified reactor disassembly basins, or the potential use of the BNFP. The reactor basins could be used to store the non-aluminum-based spent nuclear fuel currently in the RBOF because the poorer water quality in the basins would not cause additional corrosion for this other fuel that is not aluminum based. Recent improvements in reactor basin water chemistry control have resulted in a substantial decrease in the potential for corrosion of aluminum-clad spent nuclear fuels.

F.1.3.2 Idaho National Engineering Laboratory

The Idaho National Engineering Laboratory has several reactors and critical assemblies operating and also possesses several reactors that are either in standby or shutdown and awaiting decommissioning. From 1953 until 1992, the Idaho National Engineering Laboratory was responsible for processing and recovering highly-enriched uranium (HEU) from naval reactors. The Idaho National Engineering Laboratory discontinued processing spent nuclear fuel in 1992. Consequently, the Idaho National Engineering Laboratory has spent nuclear fuel facilities, spent nuclear fuel in storage, and spent nuclear fuel from

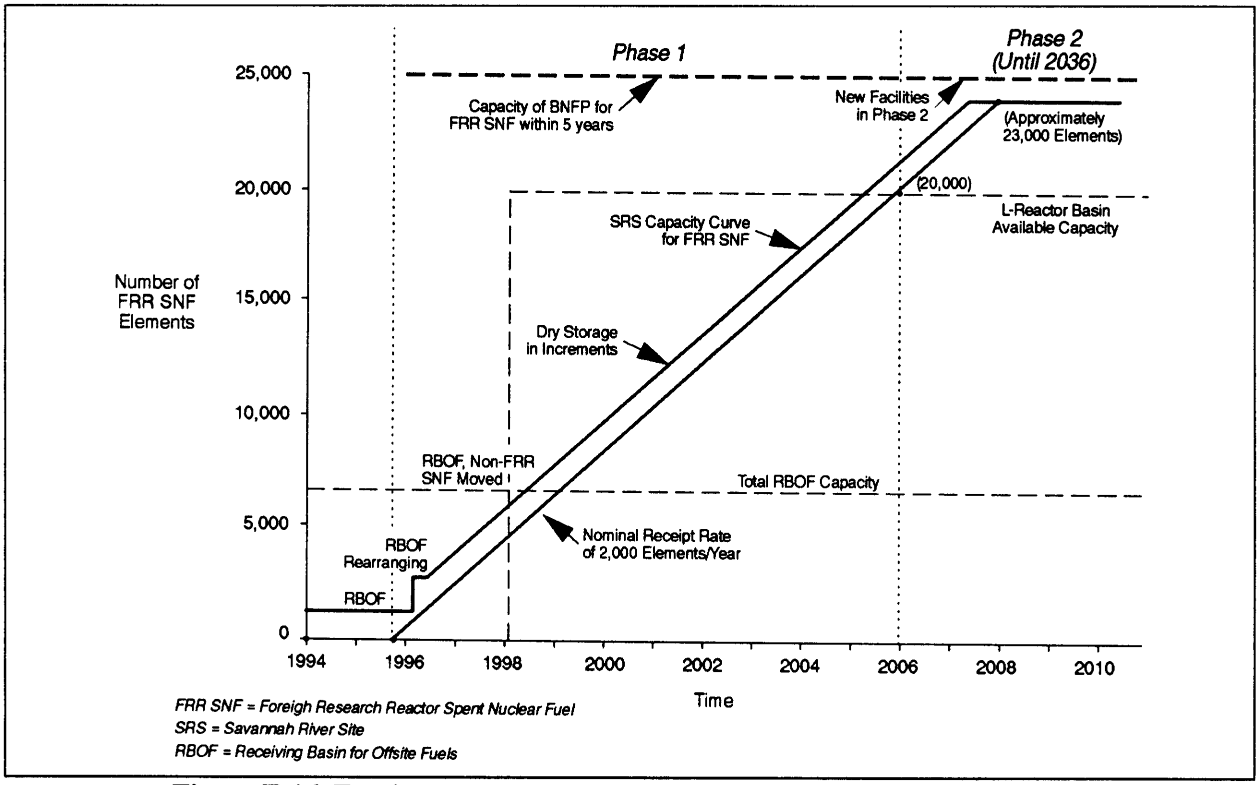


Figure F-16 Foreign Research Reactor Spent Nuclear Fuel Storage at the Savannah River Site

current operations. The Idaho National Engineering Laboratory site map with spent nuclear fuel facilities is shown in Figure F-17.

F.1.3.2.1 Spent Nuclear Fuel Activities at the Idaho National Engineering Laboratory

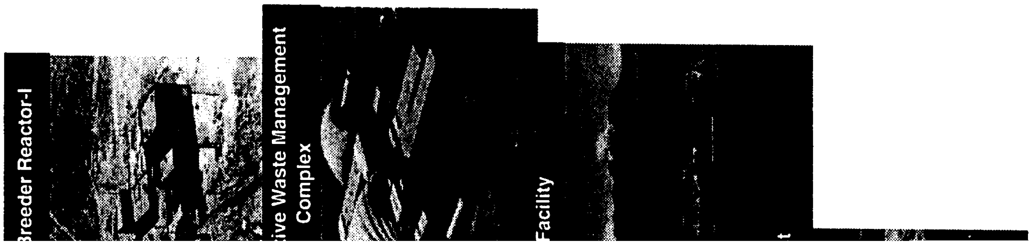
Six major facility areas at the Idaho National Engineering Laboratory store spent nuclear fuel:

- ICPP,
- Test Area North,
- Power Burst Facility,
- Test Reactor Area,
- Argonne National Laboratory-West, and
- Naval Reactors Facility.

A description of each major facility area and its spent nuclear fuel storage activities is presented below.

F.1.3.2.2 Spent Nuclear Fuel Storage Facilities at the Idaho National Engineering Laboratory

Spent nuclear fuel at the Idaho National Engineering Laboratory is sorted in a variety of dry and wet configurations. The total amount of spent nuclear fuel at the Idaho National Engineering Laboratory



the Idaho

accounts for about 10 percent (by weight of heavy metal) of the spent nuclear fuel in the DOE complex (DOE, 1995g).

Table F-11 lists the primary spent nuclear fuel storage facilities, including the type of storage configuration, capacity for foreign research reactor spent nuclear fuel receipts, and accessibility. The number, variety, and location of the wet and dry configurations currently in use at the Idaho National Engineering Laboratory are largely the result of the different purposes for the facilities (e.g., at-reactor storage, storage research and development, reprocessing, and fuel research and development). The condition of the spent nuclear fuel in storage is generally good, with the notable exception of minor amounts of fuel in the Underwater Fuel Storage Facility at the ICPP-603.

The ICPP has received spent nuclear fuel from many onsite and offsite reactors (including foreign research reactor spent nuclear fuel) for reprocessing. Reprocessing for recovery of HEU materials was ceased in 1992. The ICPP now has the mission of managing its current spent nuclear fuel inventory and assigned new spent nuclear fuel receipts, development of technologies in support of dispositioning the spent nuclear fuel, and eventually packaging the material for shipment to a repository. The ICPP stores virtually all types of spent nuclear fuel except production reactor fuel (i.e., fuel from the Hanford Site and the Savannah River Site production reactors). It stores nonproduction reactor aluminum, stainless steel, zirconium, and graphite-clad spent nuclear fuel and uses both wet and dry storage configurations. The ICPP facilities have experience and some capacity for foreign research reactor spent nuclear fuel storage. These are discussed in more detail in Section F.3.

The Test Area North has been a reactor testing facility and has received significant amounts of spent nuclear fuel for examination and testing purposes. This includes the commercial dry storage cask demonstration program and the Three Mile Island debris examination program. It has a very large hot cell and an adjacent underwater storage pool to support the testing programs. It also has a large hot shop where large pieces of equipment, such as transportation casks, have been reconfigured or maintained. At the current time, the Test Area North hot cell and pool have no future mission, but may be used by the U.S. Navy. If Test Area North is not used by the Navy, then the Test Area North hot cell and pool may have significant capacity for receipt of foreign research reactor spent nuclear fuel and for placing it into temporary underwater storage or dry storage casks.

Other storage areas such as the Power Burst Facility reactor canal and the MTR storage pool have limited storage capacities for receipt or storage of foreign research reactor spent nuclear fuel.

The Argonne National Laboratory-West facilities supported the Experimental Breeder Reactor program and also contain the Transient Reactor Test Facility, the Zero Power Physics Reactor, and the Neutron Radiography Reactor. Spent nuclear fuel storage facilities include an at-reactor molten sodium storage pool, in-process lag storage in the Hot Fuel Examination Facility and dry underground SILOs for spent fuel and wastes pending disposition. The Hot Fuel Examination Facility would be suitable for foreign research reactor spent nuclear fuel examination activities.

The Naval Reactors Facility is also located at the Idaho National Engineering Laboratory, but is not included in Table F-11 because of its sole purpose to support the Naval ship propulsion program. The Naval Reactors Facility includes the Expended Core Facility, which receives and examines Naval spent nuclear fuel to support fuel development and performance analyses. In addition, the Expended Core Facility removes structural support material from the Naval spent nuclear fuel before transfer of the fuel portion to the ICPP for reprocessing or interim storage.

Table F-11 Description of Existing Spent Nuclear Fuel Facilities at the Idaho National Engineering Laboratory

Facility	*Description*	*Capacity for Foreign Research Reactor Spent Nuclear Fuel*	*Access*
ICPP-666 Underwater Fuel Storage Area	Water Storage Facility with 6 lined storage basins 9.4 m x 14.2 m by 9.4 m or 12.5 m deep (31 ft x 46.5 ft x 31 ft or 41 ft deep)	Temporary storage after reracking for 8,400 elements	Shipment by truck. Rail shipments to a site receiving area 8 km (5 mi) away.
ICPP-603 Underwater Fuel Storage Area	Water Storage Facility with three basins of varying sizes, no sealant or liner	Not Available - facility is being shut down	Shipment by truck.
ICPP-603 Irradiated Fuel Storage Facilities	Dry Storage Facility with remote unloading area and vault storage with 636 0.5 x 3.4 m L (18 in x 11 ft long) containers	200 containers available for storage of 9,000 foreign research reactor elements	Shipment by truck. Rail shipments to a site receiving area 8 km (5 mi) away.
ICPP-749 Underground Fuel Storage Area	Dry Storage Facility with 218 underground SILOs	Approximately 60 SILOs available following renovation of first generation SILOs; capacity for 3,600 elements after fiscal year 1998	Requires receipt into ICPP-666 or ICPP-603 IFSF and packaging and conditioning for dry storage.
Test Area North-607 Pool and Hot Cell	Water Storage Facility with adjacent remote hot cell	Approximately 56 m^2 (600 ft^2) of basin 7.3 m (24 ft) deep. Capacity for 4,000 elements after new rack installation.	Additional storage space available in hot cell. Shipment by truck, cask unloading in hot cell.
Test Reactor Area-620 Power Burst Facility	Small water storage pool adjacent to Power Burst Facility	Minimal space available	Shipment by truck. Crane capacity inadequate for foreign

technologies and requirements to place DOE spent nuclear fuel in safe interim storage. Long-term activities include the development of final waste acceptance criteria requirements and stabilization technologies for alternate fuel disposition, construction of facilities to stabilize fuel to meet waste disposal requirements, processing of the fuel to a final waste form, and transportation of the waste form for disposition (discussed in more detail in Section F.3). As shown in Figure F-18, the Idaho National Engineering Laboratory has sufficient capacity for foreign research reactor spent nuclear fuel if the existing facilities are supplemented by dry casks.

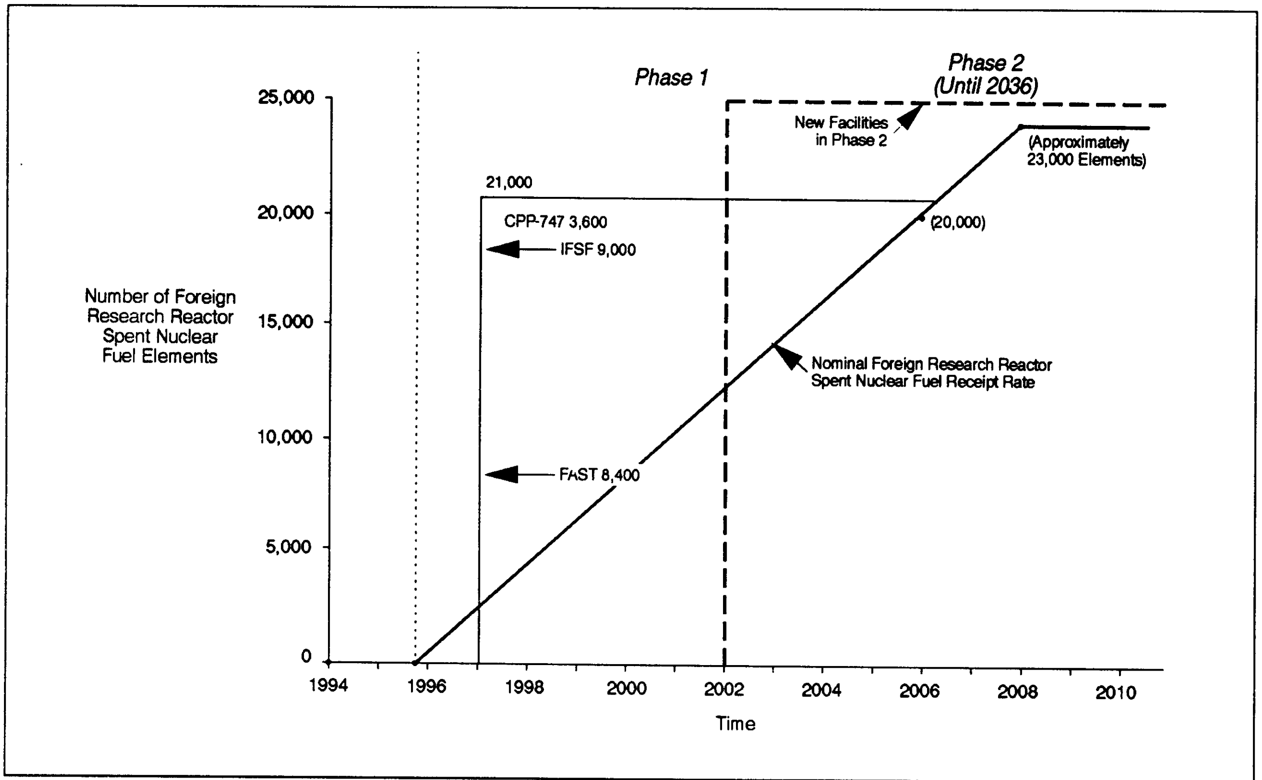


Figure F-18 Foreign Research Reactor Spent Nuclear Fuel Storage Capacity at the Idaho National Engineering Laboratory

F.1.3.3 Hanford Site

The Hanford Site lies within the semi-arid Pasco Basin of the Columbia Plateau in southeastern Washington State (DOE, 1995g). The Hanford Site occupies an area of around 1,450 km^2 (560 mi^2) north of the confluence of the Yakima and Columbia Rivers. Only about six percent of the site has been disturbed in the process of special nuclear materials production for national defense reprocessing and used for DOE purposes, such as nuclear materials production, processing, research and development, and waste management. The Hanford Site facilities [illegible]

- waste management, including new processing facilities and retrievable disposal, and
- research and development into energy, environmental, and waste management technologies.

A Tri-Party Agreement between DOE, the U.S. Environmental Protection Agency, and the State of Washington provides milestones and guidance for these activities at the Hanford Site. Current schedules use a 2030 date for the completion of most of the restoration activities at the site. A map of the Hanford Site that shows spent nuclear fuel facilities is presented in Figure F-19. Existing spent nuclear fuel facilities are listed in Table F-12.

F.1.3.3.1 Spent Nuclear Fuel Activities at the Hanford Site

The following spent nuclear fuel types and their associated facilities are at the Hanford Site:

- *N Reactor Spent Nuclear Fuel:* This is zircaloy-clad, metallic uranium fuel stored in water in the 105-KE and 105-KW Basins (1,146 and 954 MTHM, respectively), and exposed to air in the plutonium-uranium extraction dissolver cells A, B, and C (0.3 MTHM).
- *Single-Pass Reactor Spent Nuclear Fuel:* This is aluminum-clad, metallic uranium fuel stored in water in the 105-KE and 105-KW Basins (0.4 and 0.1 MTHM, respectively), and stored in water in the plutonium-uranium extraction basin (approximately 2.9 MTHM).
- *Fast Flux Test Facility Spent Nuclear Fuel:* This consists of stainless steel-clad fuel stored in liquid sodium at the Fast Flux Test Facility, comprised mainly of a uranium/plutonium oxide fuel, but with some carbide, metallic, and nitride fuel elements (in all, fuel from 329 assemblies of spent nuclear fuel).
- *Shippingport Core II Spent Nuclear Fuel:* These assemblies are zircaloy-clad uranium dioxide fuel, and are stored in the T-Plant Canyon, Pool Cell 4.
- *Miscellaneous Commercial and Experimental Spent Nuclear Fuel:* This includes primarily zircaloy-clad uranium dioxide fuel stored in air, but does include some Test, Research, Isotope, General Atomic (TRIGA) reactor hydride spent nuclear fuel stored in water and aluminum-clad, uranium-aluminum metallic fuel stored in air. These are principally stored in the 300-Area at Hanford Site.

F.1.3.3.2 Spent Nuclear Fuel Storage Facilities at the Hanford Site

The Hanford Site spent nuclear fuel storage facilities are principally based upon wet methods. Table F-12 provides a brief summary of these facilities. The age, condition, available capacity of these facilities, and the Tri-Party Agreement milestones generally prevent the use of the existing facilities for storage of foreign research reactor spent nuclear fuel. It is extremely unlikely that significant processing activities on spent nuclear fuel will occur in the near future, and thus new facilities would be required for foreign

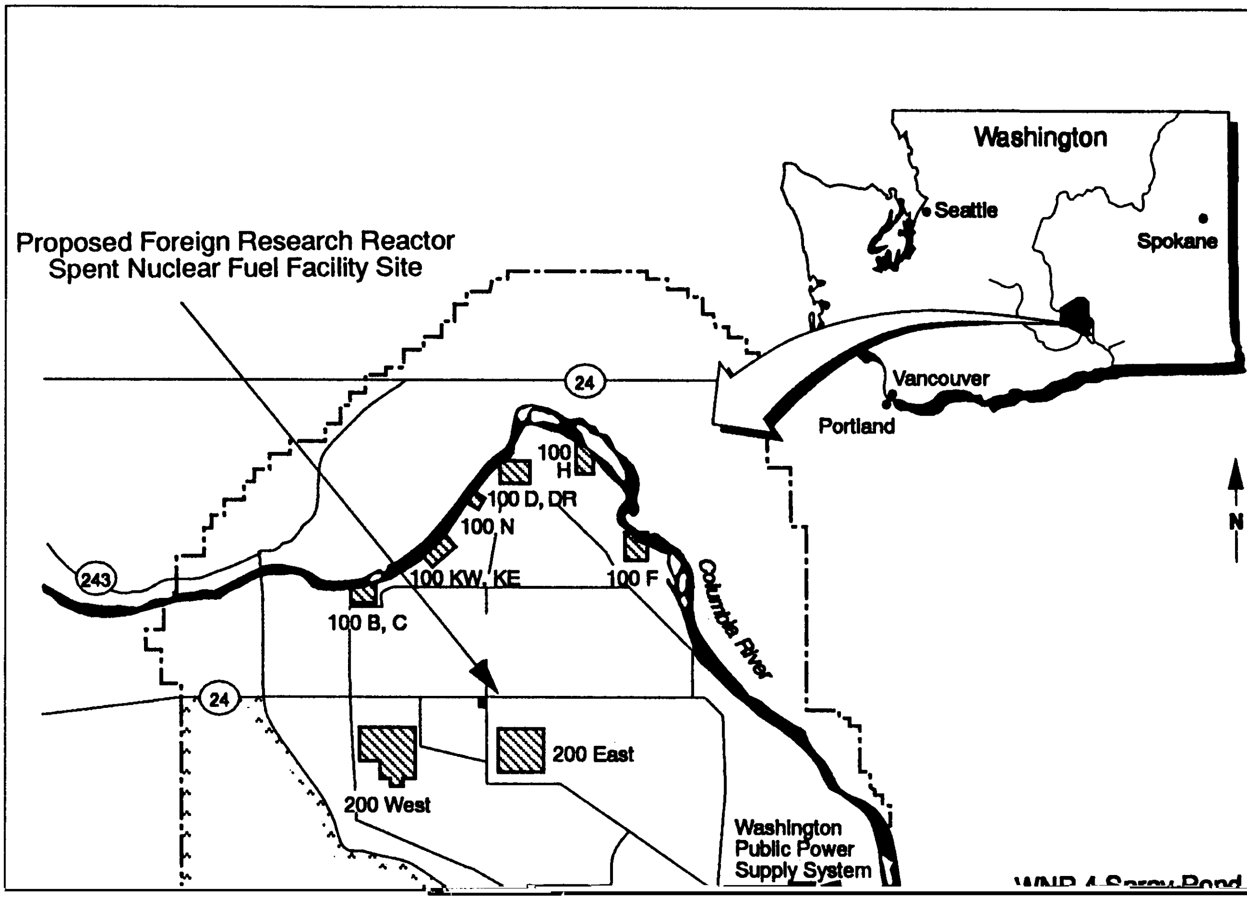

Washington
Seattle
Spokane
Vancouver
Portland
Proposed Foreign Research Reactor
Spent Nuclear Fuel Facility Site
24
243
100 H
100 D, DR
100 N
100 KW, KE
100 F
100 B, C
Columbia River
N
200 East
200 West
Washington
Public Power
Supply System

Table F-12 Description of Existing Spent Nuclear Fuel Facilities at Hanford Site

		Capacity for Foreign Research Reactor Spent	

specifies that Hanford generated spent nuclear fuel will remain in storage at Hanford pending decisions on ultimate disposition. The second, is the Management of Spent Nuclear Fuel from the K Basins at the

addresses the location and method of managing Hanford spent nuclear fuel for up to 40 years or until

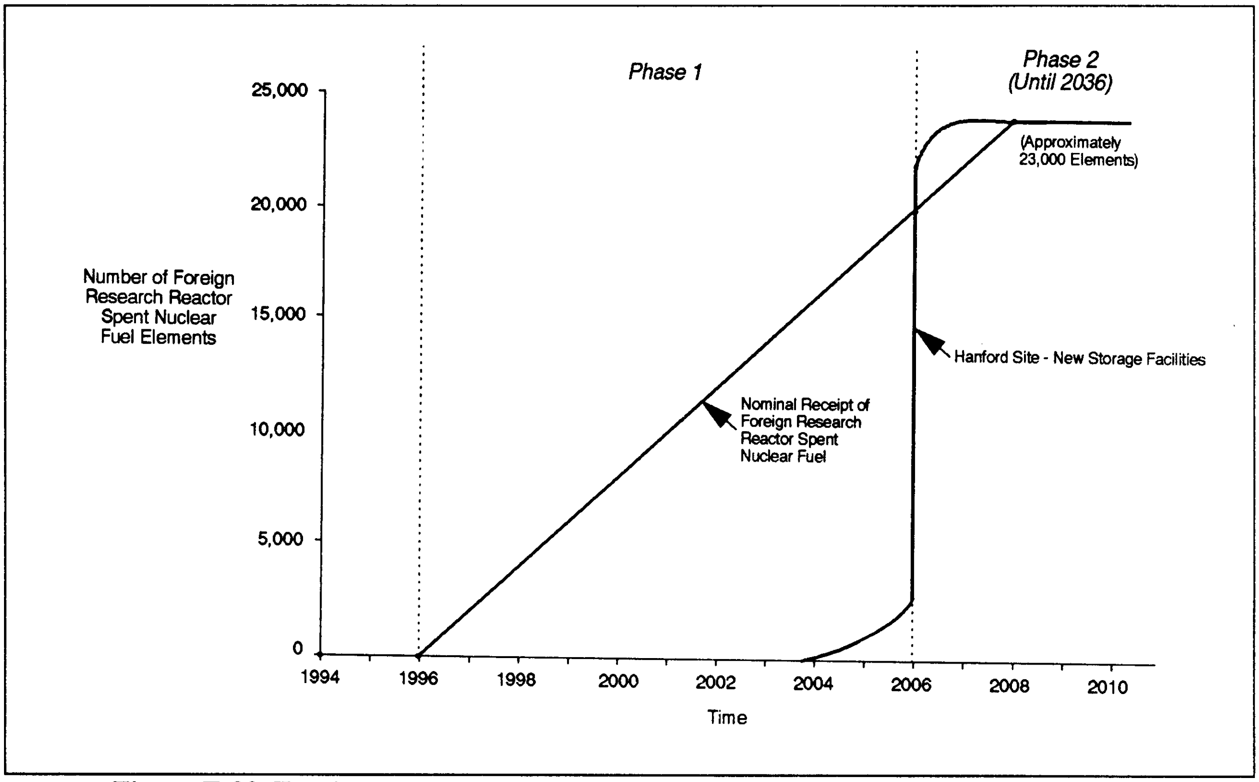


Figure F-20 Foreign Research Reactor Spent Nuclear Fuel Storage Capacity at the Hanford Site

requirements. A map of the Oak Ridge Reservation and its candidate sites for foreign research reactor spent nuclear fuel storage is presented in Figure F-21.

F.1.3.4.1 Spent Nuclear Fuel Activities at the Oak Ridge Reservation

Most Oak Ridge Reservation spent nuclear fuel activities occur at the Oak Ridge National Laboratory. The Oak Ridge National Laboratory has operated several small research reactors, all of which generate (or have generated) spent nuclear fuel. These reactors all have small fuel preparation and handling facilities associated with them ranging up to the single digit MTHM capacity. The spent nuclear fuel storage space is small, and most is either full or committed, with little excess capacity. The Oak Ridge National Laboratory also has hot cell and irradiated fuel examination facilities. Currently, only the High Flux Isotope Reactor is operating and generating spent nuclear fuel. More spent nuclear fuel facilities at Oak Ridge Reservation are presented in Table F-13.

F.1.3.4.2 Spent Nuclear Fuel Storage Facilities at the Oak Ridge Reservation

The Oak Ridge Reservation stores spent nuclear fuel in several small facilities. Most of these facilities are old and are unlikely to meet modern building code and seismic standards. The spent nuclear fuel facilities include the following structures:

- *Building 3525* ...

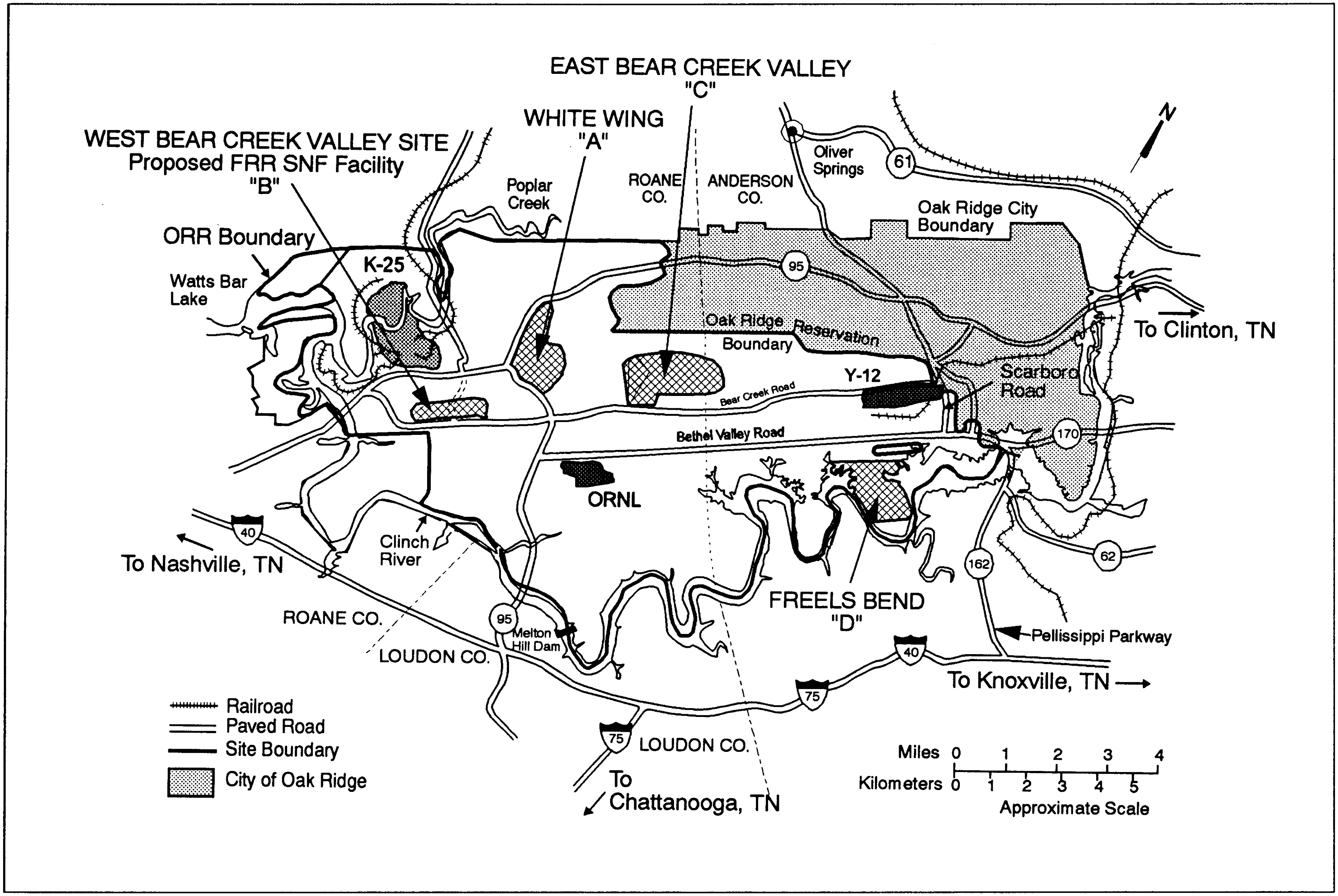


Figure F-21 Candidate Sites at the Oak Ridge Reservation for Foreign Research Reactor Spent Nuclear Fuel Storage

Table F-13 Major Spent Nuclear Fuel Facilities at the Oak Ridge Reservation

Facility	Characteristics	Capacity for Foreign Research Reactor Spent Nuclear Fuel	Access
Building 3525	Hot Cells	No, too small	Truck
Building 4501	Hot Cells	No, too small	Truck
Building 7827	Drywells	No space	Truck
Building 7920	Hot Cells	No, too small	Truck
Building 9720-5 (Y-12)	Warehouse	No, unirradiated fuel only	Truck
Other	Research Reactors	No, storage space near capacity	Truck

- *Building 4501 - High-Level Radiochemical Facility:* This facility dates from 1951 and contains hot cells for examining irradiated materials. This facility contains small quantities (several kg) of sectioned commercial fuel.

- *Building 7920 - Radiochemical Engineering Development Center:* This is a multi-purpose, hot cell facility for (relatively) large quantities of irradiated spent nuclear fuel. This facility supports target preparation and processing for the High Flux Isotope Reactor and contains samples and targets of research reactor spent nuclear fuel in dry storage.

- *Building 9720-5 (Y-12):* This is a large warehouse for storing and safeguarding unirradiated or low burnup HEU fuel. It currently contains around 0.2 MTHM.

- *Research Reactors:* There are five existing and one planned research reactor at the Oak Ridge Reservation. All of these reactors have small spent nuclear fuel storage basins nearby, and this capacity is essentially full. Only the High Flux Isotope Reactor is currently operating.

- *The Oak Ridge Reservation* also has several drywells such as Building 7827 and drum storage areas for irradiated fuel. Spent nuclear fuel would be relocated in accordance with actions of the DOE Programmatic SNF&INEL Final EIS (DOE, 1995g).

None of these locations have any significant capacity for the potential quantities of foreign research reactor spent nuclear fuel.

F.1.3.4.3 Planned or Potential Spent Nuclear Fuel Storage Facilities at the Oak Ridge Reservation for Foreign Research Reactor Spent Nuclear Fuel

The Oak Ridge Reservation has plans for dry storage of spent nuclear fuel. This would be accomplished via a modular route at the High Flux Isotope Reactor location. This dry storage area could be extended almost indefinitely to accommodate the Oak Ridge Reservation's needs.

DOE is evaluating a spent nuclear fuel management complex for handling DOE spent nuclear fuel from other sites as an alternative in the DOE Programmatic SNF&INEL Final EIS (DOE, 1995g). The spent nuclear fuel management complex would include the following:

- Spent Nuclear Fuel Receiving and Canning Facility,

- Technology Development Facility,

- Interim Dry Storage Facility, and

- Expended Core Facility for Naval-type fuel similar to the one at Idaho National Engineering Laboratory.

The receiving and canning facility would receive spent nuclear fuel cask shipments from offsite and prepare the spent nuclear fuel for dry storage. The facility incorporates a pool (wet) storage facility for cooling spent nuclear fuel (tentatively identified as a 5-year period) prior to placement into dry storage, as necessary. The technology development facility would investigate the applicability of dry storage technologies and pilot scale technology development for disposal for various types of spent nuclear fuel. The interim dry storage area would consist of passive storage modules to safely store the spent nuclear fuel for 40 years. Naval fuel would be examined at the Expended Core Facility prior to interim storage. The total land required for the facility, including a buffer zone, is approximately 36 ha (90 acres).

The proposed site for the spent nuclear fuel facilities is located in the West Bear Creek Valley Area, in the western portion of the Oak Ridge Reservation site. This area of the Oak Ridge Reservation is currently in the Natural Areas land use category and is designated for future Waste Management land use. Land uses bordering on the Oak Ridge Reservation in this area are primarily agricultural farmland and commercial forest, with sparsely located residences (i.e., low population density).

Environmental, safety, and health consequences are calculated to be negligible from the spent nuclear fuel facilities, although a preliminary design and/or layout is not provided. Releases of krypton-85, chlorine, and hydrogen fluoride are included in the analysis for incident-free operations, but the source of these emissions is not reported. Facility budgetary requirements are not delineated.

Foreign research reactor spent nuclear fuel represents less than one percent of the DOE spent nuclear fuel quantities in terms of mass and, thus, its effect would be minimal as compared to the other fuels. The foreign research reactor spent nuclear fuel contribution to the operational consequences and its costs are not delineated. Figure F-22 summarizes foreign research reactor spent nuclear fuel capacity at the Oak Ridge Reservation. New facility construction would be required for foreign research reactor spent nuclear storage.

F.1.3.5 Nevada Test Site

The Nevada Test Site is located in the southeastern part of the State of Nevada, and is used as the on-continent site for nuclear weapons testing (DOE, 1995g). The Nevada Test Site encompasses approximately 3,500 km^2 (1,350 mi^2) of desert land, with flats, mesas, and mountain ridges (Figure F-23). Essentially no permanent surface waters exist, and the depth to groundwater routinely exceeds 330 m (1,000 ft). The Nellis Air Force Base Range surrounds the Nevada Test Site to the north, east, and west; and, with the Tonapah Test Range, provides a 24 to 104 km (15 to 65 mi) buffer zone between the Nevada Test Site and public lands. The Bureau of Land Management owns land on the southern and southwestern borders of the Nevada Test Site. Principal access to the site is via the town of Mercury, on the southeastern corner. Las Vegas is approximately 104 km (65 mi) from this corner of the Site.

Activities at the site have included nuclear weapons testing, nuclear reactor tests, nuclear rocket engine development, and waste management. Current activities include nuclear weapons-related activities (e.g., emergency search teams, arms control/verification, etc.), low-level waste/low-level mixed waste disposal, and site characterization for commercial spent nuclear fuel disposal.

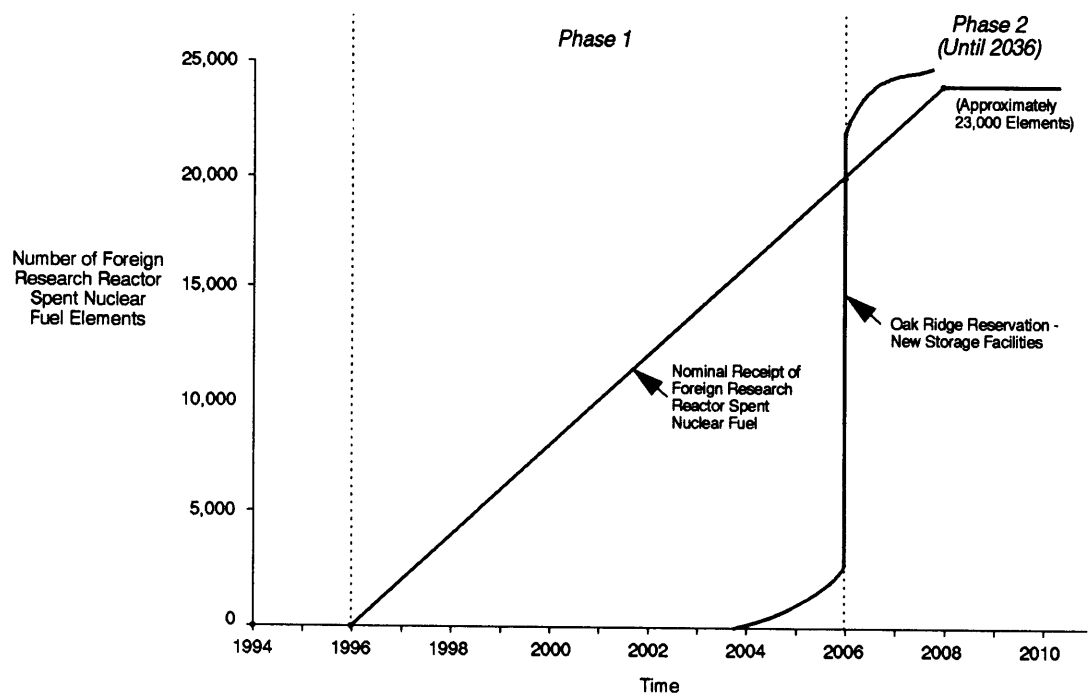


Figure F-22 Foreign Research Reactor Spent Nuclear Fuel Storage Capacity at the Oak Ridge Reservation

F.1.3.5.1 Spent Nuclear Fuel Activities at the Nevada Test Site

The Nevada Test Site has several existing facilities that could be useful for spent nuclear fuel management. These facilities were principally used for nuclear rocket engine development and are located at Jackass Flats, in a southern portion of the Nevada Test Site called the Nevada Research & Development Area (Cosimi, 1994; Chandler et al., 1992; Gertz, 1994; Hynes, 1994; Reed, 1994). The facilities include several large hot cell and fuel examination "shops," with large cranes and manipulators. At least two of these facilities appear to be ideally suited for handling and storing foreign research reactor spent nuclear fuel after relatively minor upgrades and refurbishments. Table F-14 summarizes the capabilities of these facilities for foreign research reactor spent nuclear fuel.

The Engine Maintenance and Disassembly (E-MAD) facility was originally constructed for the assembly and preparation of nuclear rocket engines for testing, refurbishment of activated engines, and disassembly and inspection of tested engines and components. The facility is designed for remote handling and examination of highly radioactive components. The building is a T-shaped, multi-storied structure, with overall dimensions of 85 x 107 m (280 ft x 350 ft) (Figure F-24). Numerous hot cells exist, with remote handling and transfer equipment, and the largest hot cell is 20 m wide x 45 m long x 23.5 m high (66 ft wide by 146 ft long and 77 ft high). Typically, 1.5 m (5 ft) thick concrete walls provide the shielding. Material transfer capabilities include several 36 metric tons (40 ton) cranes and a cask handling system of

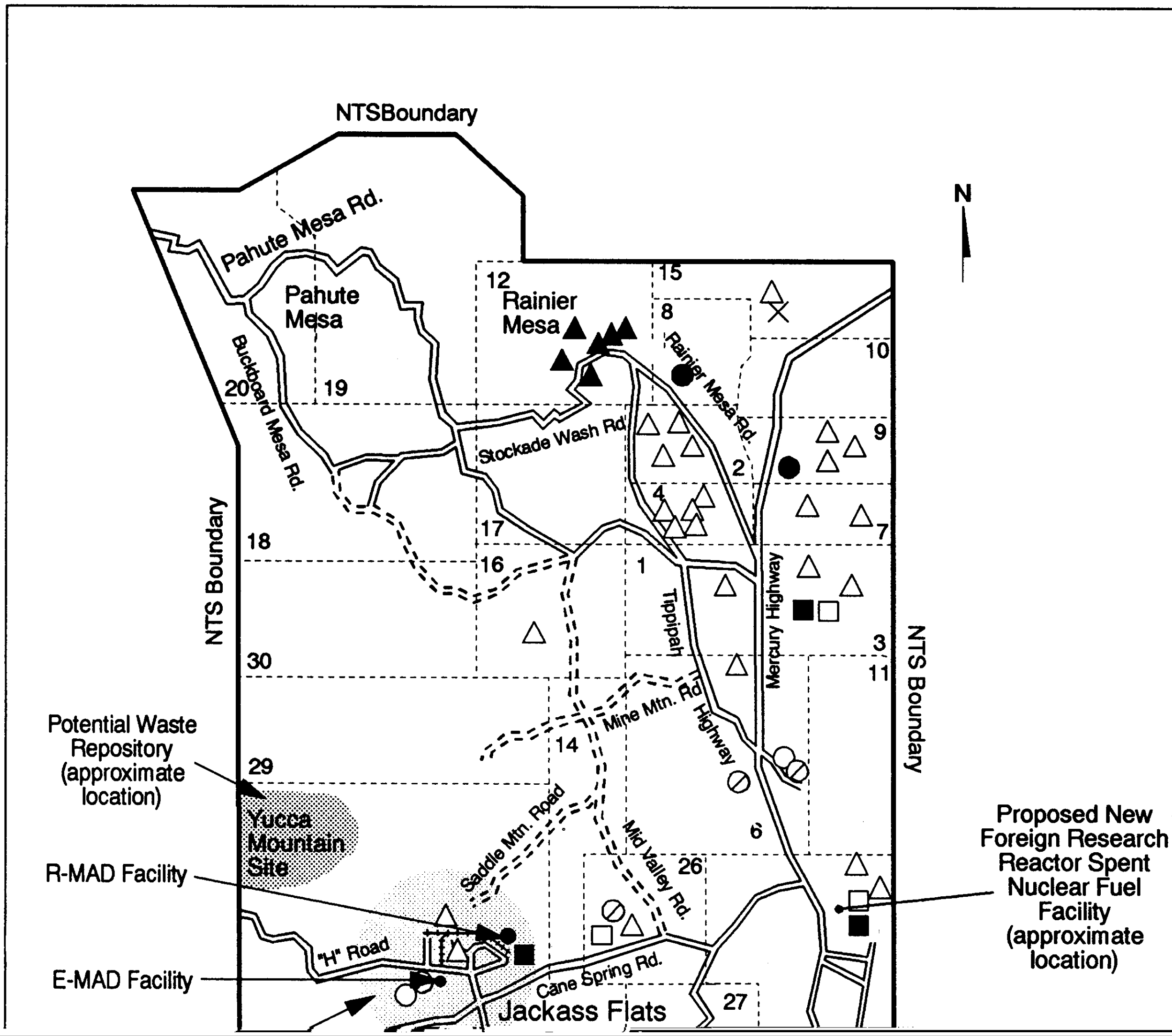

NTSBoundary
N
Pahute Mesa Rd.
Pahute
Mesa
Buckboard Mesa Rd.
20
19
12
Rainier
Mesa
15
8
Rainier Mesa Rd.
10
9
Stockade Wash Rd
2
4
7
18
17
16
1
Tippipah
Mercury Highway
3
11
NTS Boundary
30
Mine Mtn. Rd
Highway
14
29
Potential Waste
Repository
(approximate
location)
Yucca
Mountain
Site
Saddle Mtn. Road
Mid Valley Rd.
6
26
R-MAD Facility
"H" Road
Cane Spring Rd.
E-MAD Facility
Jackass Flats
27
NTS Boundary
Proposed New
Foreign Research
Reactor Spent
Nuclear Fuel
Facility
(approximate
location)

Table F-14 Major Spent Nuclear Fuel-Capable Facilities at the Nevada Test Site

Facility	Characteristics	Capacity for Foreign Research Reactor Spent Nuclear Fuel	Access
E-MAD	Large hot cell, with smaller hot cells. Main hot cell area is 895m^2 (9,600 ft^2)	Yes, as either a vault or a staging facility for dry casks, after 1-2 years refurbishments, 25,000 elements	Truck
R-MAD	Large hot cell, with smaller hot cells. Main hot cell area is 223m^2 (2,400 ft^2)	Yes, as a staging facility for dry casks or a small vault, after 1-3 years refurbishment, 25,000 elements	Truck

approximately 91 metric tons (100 tons) capacity. The heating, ventilation, and air conditioning systems for the hot cell areas maintain negative pressure and exhaust through High Efficiency Particulate Air filters.

The E-MAD facility is currently unused and last saw service during the 1980s for commercial spent nuclear fuel storage experiments (e.g., Climax Mine Project) (Gertz, 1994). Thirteen commercial spent nuclear fuel assemblies were tested in casks and drywells. The E-MAD facility was subsequently used to load transportation casks for shipment of the spent nuclear fuel to Idaho. Several of these spent nuclear fuel storage casks remain at the site (Hynes, 1994). The Los Alamos National Laboratory assessment (Chandler et al., 1992) considers the facility to require only minor upgrades and routine maintenance.

The Reactor Maintenance and Disassembly facility is located a short distance from the E-MAD facility. This facility contains two (contact) assembly bays and one remotely operated hot disassembly bay. The hot bay dimensions are 18 x 12 x 18 m (60 by 40 by 60 ft) high, with 1.8 m (6 ft) thick walls for shielding. A transfer system connects six hot cells to the hot disassembly bay. The Los Alamos National Laboratory assessment (Chandler et al., 1992) found the Reactor Maintenance and Disassembly facility to require a minor upgrade.

F.1.3.5.2 Spent Nuclear Fuel Storage Facilities at the Nevada Test Site

At the present time, the Nevada Test Site is not storing spent nuclear fuel. As noted, facilities in the Jackass Flats area have handled spent nuclear fuel in the past and could be adapted to accommodate

foreign research reactor spent nuclear fuel and serve as the nucleus of a spent nuclear fuel storage facility. The E-MAD and Reactor Maintenance and Disassembly facilities appear to have sufficient size and design for accommodating all of the foreign research reactor spent nuclear fuel in a dry storage mode, either vault

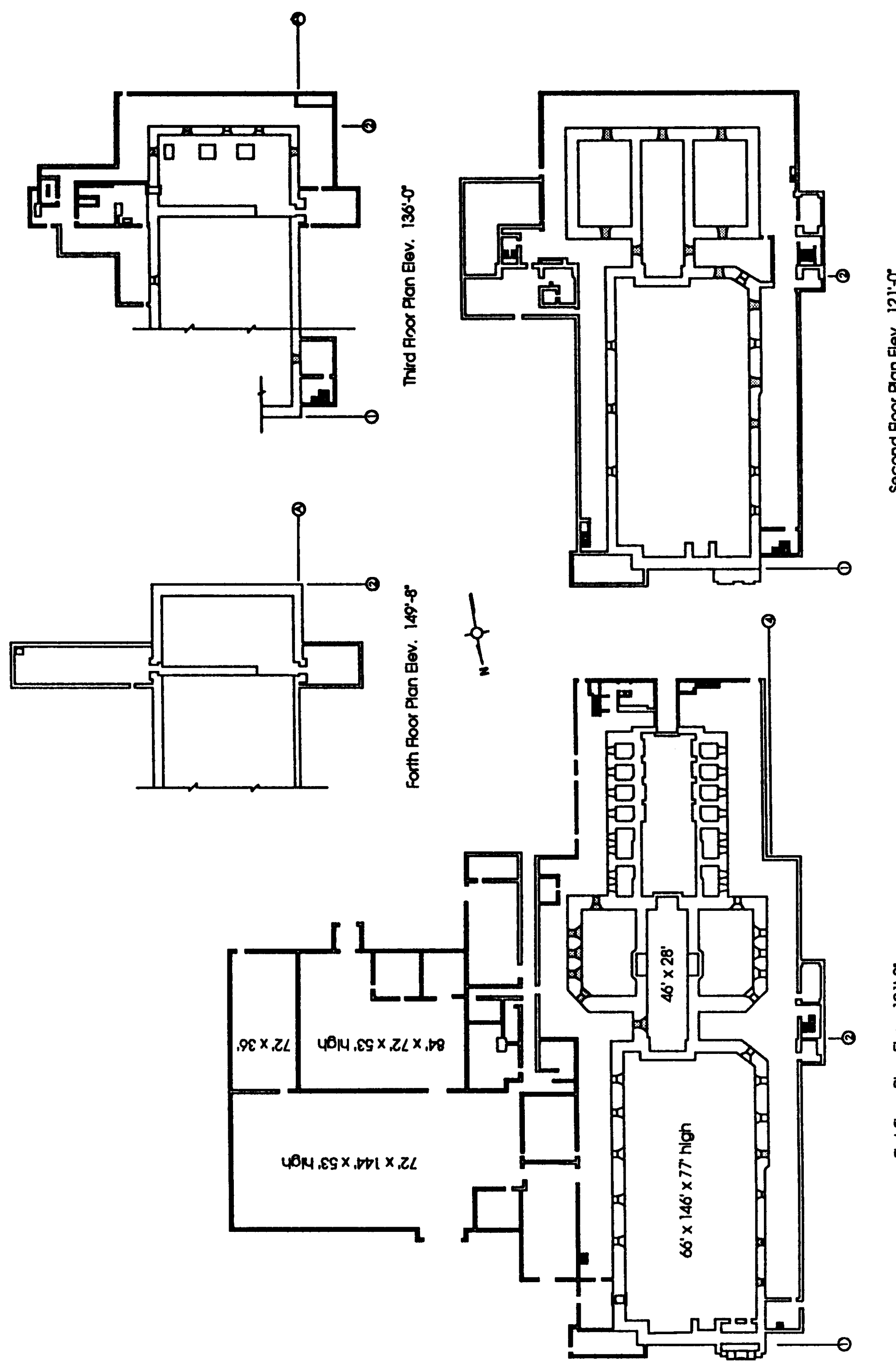


Figure F-24 Schematic of the Engine Maintenance and Disassembly Facility at the Nevada Test Site

The receiving and canning facility would receive spent nuclear fuel cask shipments from offsite and prepare the spent nuclear fuel for dry storage. The facility incorporates a pool (wet) storage facility for [illegible]

necessary. The technology development facility would investigate the applicability of dry storage technologies and pilot scale technology development for disposal for various types of spent nuclear fuel. The interim dry storage area would consist of passive storage modules to safely store the spent nuclear fuel for 40 years. Naval fuel would be examined at the Expended Core Facility prior to interim storage. The total land required for the facility, including a buffer zone, is approximately 36 ha (90 acres).

Environmental, safety, and health consequences are calculated to be negligible from the spent nuclear fuel facilities, although a preliminary design and/or layout is not provided. Releases of krypton-85, chlorine, and hydrogen fluoride are included in the analysis for incident-free operations, but the source of these emissions is not reported. Facility budgetary requirements are not reported.

for foreign research reactor spent nuclear fuel discussed in Section F.3. Alternatively, new facilities could be built, but these would require a longer transportation path to the proposed Yucca Mountain repository.

F.1.3.6 Storage at Overseas Facilities

Currently, foreign research reactor spent nuclear fuel is being stored in wet pools at foreign research reactor sites. These pools are approaching the levels of their capacity, which is why the foreign research reactor operators would like the United States to accept their spent nuclear fuel. An alternative being considered by DOE is foreign research reactor spent nuclear fuel storage at overseas facilities. Several facilities exist in Europe for contractual storage of both commercial and research reactor spent nuclear fuel for a fee, including:

- British facilities at Dounreay, Scotland and Sellafield, England. The former has several small pools for research reactor fuels, while the latter has several large pools with a capacity of 3,000 MTHM for commercial spent nuclear fuel (Bonser, 1994).
- French facilities at La Hague, with several large pools having a total capacity of 14,000 MTHM for commercial spent nuclear fuel (Nuclear Fuel, 1993); facilities at Marcoule, for research and metallic spent nuclear fuel.

Electricite De France has also announced its intention of constructing a commercial spent nuclear fuel wet storage facility with a capacity of 12,000 MTHM (Nuclear Fuel, 1994b). Dry storage of spent nuclear fuel is also being considered.

These facilities are predominantly stainless-steel lined wet storage pools that meet modern seismicity and confinement standards and maintain good water chemistry. Wet storage pools designed for commercial spent nuclear fuel could, after license modification and new rack installation, store foreign research reactor spent nuclear fuel. These overseas wet storage pools are similar in design and layout to the generic wet storage facility discussed in Section F.3.

F.1.4 Vitrified Waste Storage Facilities

If foreign research reactor spent nuclear fuel is processed, the resulting high-level waste would be vitrified and placed into stainless steel canisters. The Savannah River Site is the only domestic site that currently has a storage facility designed and built for storing vitrified high-level waste from the processing of spent nuclear fuel. This facility is termed the Glass Waste Storage Building, and it is located immediately adjacent to the Savannah River Site vitrification facility (the Defense Waste Processing Facility), in the S-Area of the site near the H-Area processing facilities (DOE, 1994g). Figure F-26 provides a general overview of the facilities in the S-Area. Figure F-27 displays a general layout of the building. The Glass Waste Storage Building is designed to accommodate the standard Defense Waste Processing Facility vitrified waste canister (Figure F-28). The existing building has space for 2,286 of these canisters. A second, almost identical building, is planned for construction starting in 2007. Additional buildings may be built, up to a total interim storage capacity of 10,000 canisters if delays in the Federal Repository Program are encountered (DOE, 1994g). The Defense Waste Processing Facility/Glass Waste Storage Building area does not currently include a cask receiving/shipping facility, but one is planned for future construction.

The facility is relatively simple in design and operation. It consists of a structure enclosing a concrete floor that functions as the charging face to the vault beneath it. Shield plugs are removed from the floor to provide access to storage tubes in the vaults that would contain the canisters. Each storage tube contains

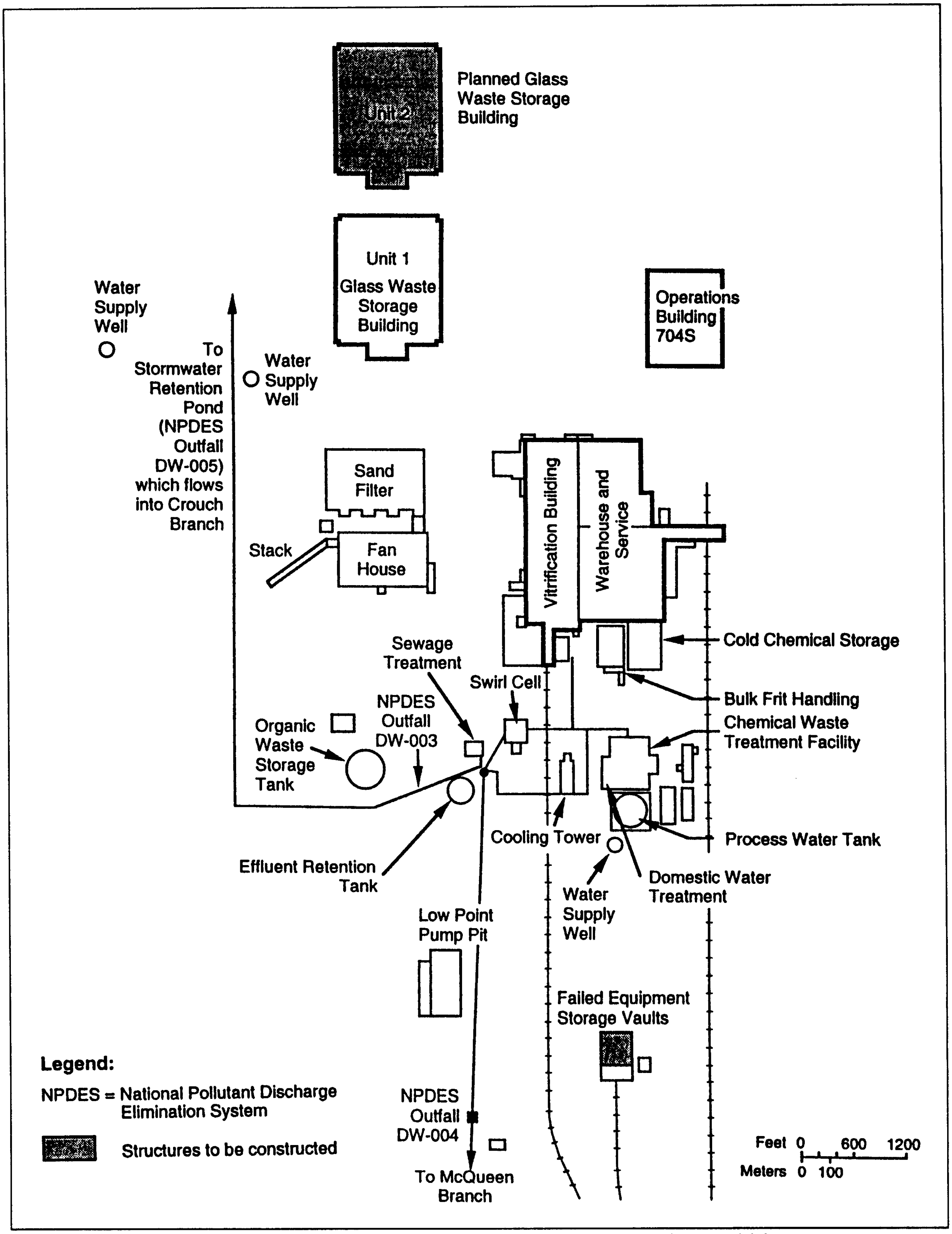


Figure F-26 General Layout of the Existing Vitrification Facilities at the Savannah River Site

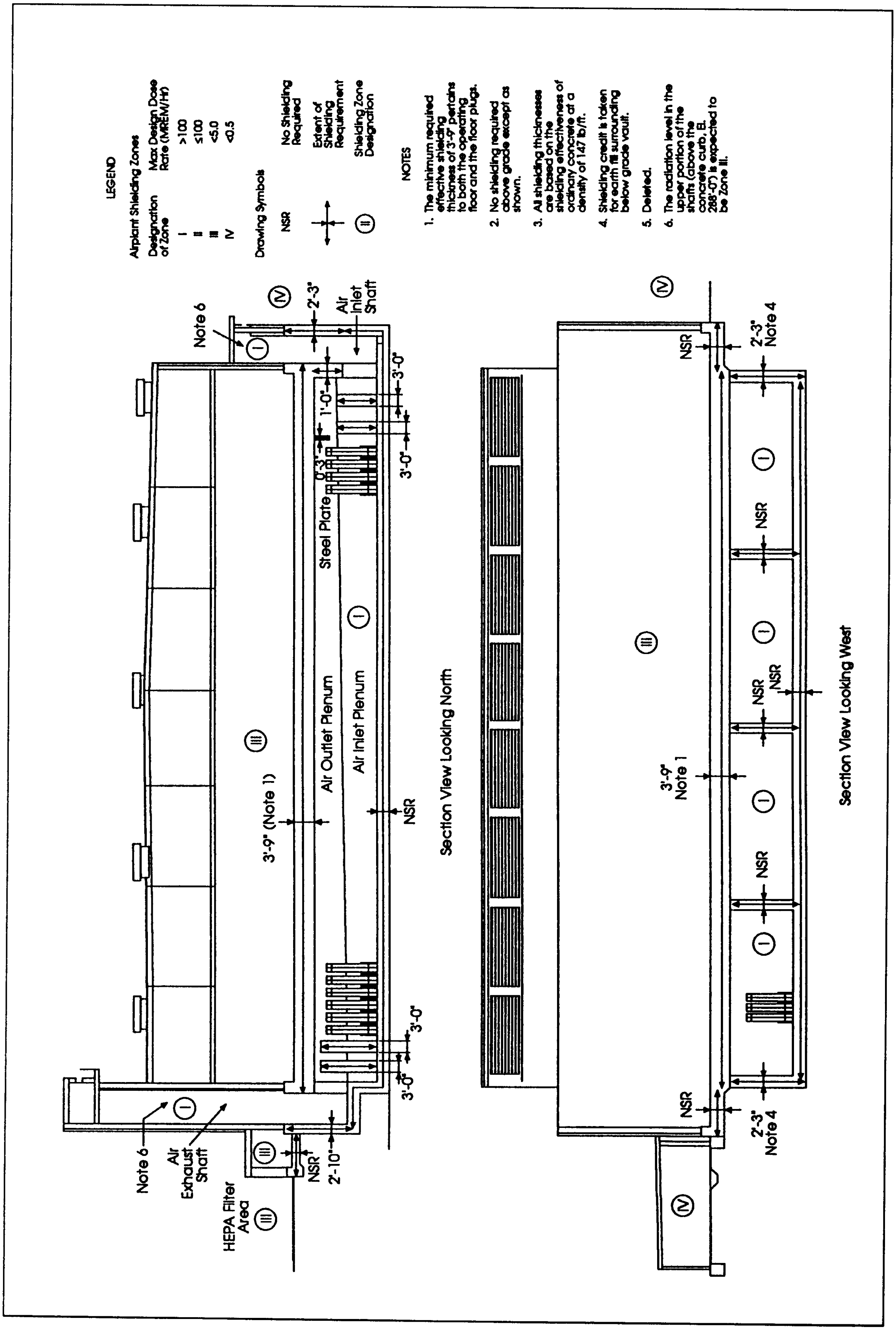
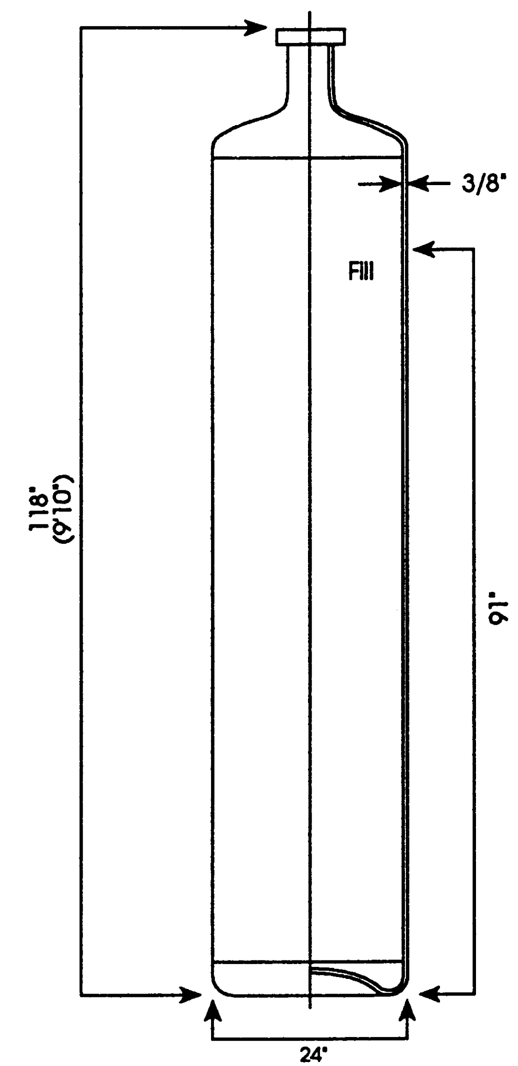


Figure F-27 Layout of the Glass Waste Storage Building



Material:	304L Stainless Steel
Empty Weight:	1,100 lb
Net Weight:	3,540 lb (Glass)
Weight of Radionuclide:	41 kg (90 lb)
Activity:	8.3×10^{15} Bq (223,000 Ci)
Decay Heat:	660 W
Radiation Field (at Surface):	6,300 rad/hr
Surface Contamination:	Less than 10^{-4} µCi/cm^2

Figure F-28 Typical Defense Waste Processing Facility Glass Waste Canister

two canisters, stacked vertically. The vault area consists primarily of steel-reinforced concrete and is designed to resist all earthquake and severe weather incidents.

Radioactive decay heat from the canisters is removed by the Glass Waste Storage Building's forced air fan exhaust system. The exhaust air is drawn around the canisters and then exhausted through the building's High Efficiency Particulate Air filtered ventilation system and discharged to the atmosphere via a stack. No condensate is expected to form, although the building does include a sump for exhaust air condensate. No radioactivity is expected in the exhaust air or in any condensates that might form.

During operation, a special dedicated transporter vehicle moves the canister from the Defense Waste Processing Facility vitrification building to the Glass Waste Storage Building in a shielded transporter. The transporter's cask is placed over the appropriate vault borehole, the shield plug is removed, and the canister is lowered via a crane mechanism into the borehole. The shield plug is replaced, and the transporter returns to the vitrification plant for the next shipment.

Several overseas facilities also exist for vitrified waste storage at Marcoule (France), La Hague (France), and Sellafield (England) (COGEMA, 1994a and 1994b; BNFL, 1994a and 1994b). These facilities are designed as natural circulation vaults and, thus, do not require fans for storage cooling. These vaults use a smaller canister, with several thousand currently in storage.

F.2 Storage Technology Evaluation Methodology

The selection of a spent nuclear fuel storage technology for foreign research reactor spent nuclear fuel requires a multi-disciplinary approach including the evaluation of, at the minimum, the environmental impact of alternatives and the following key design and performance areas:

- chemical compatibility,
- subcriticality assurance,
- shielding effectiveness,
- structural integrity (i.e., containment),
- thermal performance,
- ease of use,
- cost, and
- regulatory basis and licensing.

Other factors that may affect the decision process are whether the design has been previously licensed and actually used to store spent nuclear fuel, and its perceived ability to meet applicable regulations and standards if it has not yet been licensed.

Two principal types of spent nuclear fuel storage can be used for foreign research reactor spent nuclear fuel, wet and dry. Wet storage denotes the immersion of fuel in a pool of water, which performs the dual functions of shielding/leaking radionuclide removal and decay heat removal, but which relies on active systems. Dry storage encompasses a wide spectrum of structures that house the fuel in a dry inert gas environment, with an emphasis on passive system design and operation.

F.2.1 Chemical Compatibility

The most important criterion in assessing foreign research reactor spent nuclear fuel storage technologies is the compatibility of the spent nuclear fuel with the fuel storage technology environment. The research reactor fuel cladding is either aluminum or stainless steel. Aluminum cladding fuel is the predominant type in the mix of foreign research reactor spent nuclear fuel being considered for acceptance in this EIS. The selected method of storage for any foreign research reactor spent nuclear fuel that may be accepted must provide a benign and noncorrosive environment for the fuel.

In reviewing the corrosive potential of aluminum, acidic, alkaline, and even many neutral chemical solutions have been found to be significantly corrosive. Therefore, the use of wet storage technology for the majority of the foreign research reactor spent nuclear fuel that contains aluminum cladding would require the maintenance of high water purity throughout the storage life of the pool, which may equal or exceed 40 years.

Unlike wet storage, most of the dry storage technologies utilize a dry inert gas atmosphere for the fuel, which is a noncorrosive noble gas that also enhances conduction heat transfer from the fuel to the encapsulating container. Some dry storage technologies use dry nitrogen instead of inert gas. Even in the event of a loss of inert gas atmosphere, the air atmosphere would be less corrosive than a less-than-high-purity water pool. Finally, previous experience at many DOE wet storage facilities has shown that poor water quality dramatically deteriorates the integrity of aluminum fuel. Thus, the chemical compatibility criterion indicates that dry storage is a more appropriate technology than wet storage for extended storage of foreign research reactor spent nuclear fuel.

F.2.2 Subcriticality Assurance

Uranium and plutonium are the principal elements that have the unique ability to split or fission after absorbing a neutron, and release energy and several new neutrons from this fission process. The particular forms or isotopes of uranium that are effective in the fission process are called fissile materials and are ^{233}U, ^{235}U, and ^{239}Pu. Of these three isotopes, only ^{235}U exists naturally, while the other two isotopes can be produced artificially. Under the right conditions, the fission process can be self-sustaining or even grow by a chain reaction. This chain reaction produces as many or more neutrons than are absorbed in an assembly of fissile materials.

In nuclear engineering terminology, the numerical measure of a mass of fissile material to achieve and maintain a self-sustaining fission chain reaction is termed K-effective. K-effective is the net ratio of neutrons produced per neutron absorbed in the fissile material mass. When K-effective equals 1.0, the mass is said to be critical because it can maintain the fission process. When K-effective is less than 1.0, the mass is considered to be subcritical.

Subcriticality can be ensured by a number of factors, including:

- diluted concentration of fissile materials,
- adequate separation distance between masses of fissile materials such as nuclear fuel rods or assemblies,
- presence of materials (such as boron) mixed with the fissile material that absorbs neutrons before they can be captured by the fissile material,

- exclusion of substances such as water that can encourage the absorption of neutrons by the fissile isotopes or reflect neutrons leaving the mass of fissile materials back into the mass, and

- restricting the mass of fissile material below the minimum that nature requires to initiate, maintain, and/or sustain a fission chain reaction.

In nuclear criticality safety, the principle of double contingency is used to protect against criticality. Double contingency requires that the design of any system containing fissile material use two of the aforementioned factors to prevent the onset of criticality. Criticality analyses would be required to confirm spacing and the effects of optimum moderation, as well as the different structural materials in foreign research reactor spent nuclear fuel as compared to commercial fuel (e.g., aluminum, stainless steel, hydride, inconel in foreign research reactor spent nuclear fuel as compared to zircaloy in commercial fuel).

Another important factor is that most of the products of the fission reaction are radioactive fission products that are not capable of sustaining a fission reaction. Like most elements in nature, these fission products [illegible]

can absorb neutrons, but do not produce any neutrons or energy during this absorption. Mixed with these fission products is a small amount of fissile ^{239}Pu, which was also created during the fission process. However, the sum of the remaining ^{235}U and the created ^{239}Pu is much smaller than the original quantity and concentration of ^{235}U in the new fuel. The relatively low (in comparison to its initial value) ^{235}U enrichment and ^{239}Pu present in foreign research reactor spent nuclear fuel, coupled with the presence of for shielding and their properties are well known. They are more costly than pool water and prevent visual inspection of the spent nuclear fuel, but are not prone to material loss like pool water.

In comparing shielding designs, it is important to note that most of the shielding materials have inherent limiting temperatures (i.e., maximum allowable temperature) with the exception of the steels and cast iron. These metals' temperature limits are much greater than the aluminum-based fuel cladding temperature limit. Shielding material thermal limits include both absolute values of temperature and, in the case of concrete, temperature gradients that create thermal stresses. Wet storage pool water also has a thermal limit that is the prevention of local or bulk boiling in the pool. Operation of the spent nuclear fuel pool heat removal system prevents pool water boiling, but a postulated accident in which this system is disabled requires calculation of the time before the inception of bulk pool boiling. Adequate natural convection between adjacent fuel assemblies and within storage racks prevents local nucleate boiling in any fuel flow channel.

Shielding geometry plays an important role in the determination of a dose rate profile around the storage facility. A continuous and constant thickness of shielding completely surrounding the fuel provides a relatively constant dose rate at all locations. A shielding design that is asymmetric and contains air gaps and/or varying material thickness results in hot spots and a relatively larger variation in surface dose rates. The wet pool design offers a continuous shield of water with resulting low constant dose rates throughout the pool surface. The water and pool wall, usually steel-lined concrete, also maintain a low continuous dose rate profile outside the walls.

The dry storage concrete building and concrete cask technologies rely on concrete walls for shielding with some steel internal to the walls. The need for internal airflow passages in the concrete introduces gaps in these walls. These gaps, which are labyrinths, require complex shielding analyses and typically allow a relatively larger dose rate at the air inlets and/or outlets than at the bulk concrete wall. This effect is more significant for concrete casks than for concrete buildings because the casks are more limited in the concrete thickness that is used in their shield wall. In the case of metal casks, since there are no internal air passages in the metal shield, the dose rate is relatively uniform around the surface. Different axial shielding and neutron-gamma source terms will result in different axial dose rates for the metal and concrete casks. Inground storage systems use a relatively small amount of concrete radially coupled with the surrounding earth for shielding and employ thick steel plugs for axial shielding. The hybrid metal-concrete cask design uses shielding principles similar to the concrete cask.

In comparing spent nuclear fuel storage shielding designs, the four basic technologies can be characterized as water, lead, metal, and concrete. Water and metal provide the most uniform dose rate reduction because they do not require the inclusion of labyrinth airflow passages for decay heat removal necessary for nuclear fuel being considered for acceptance in the United States will contain fission product inventories of from 1,000 to 100,000 (maximum) Ci per assembly. Therefore, the radiation source term for shielding design purposes, assuming the same number of fuel assemblies in each storage technology unit, may be significantly smaller for foreign research reactor spent nuclear fuel than for commercial fuel. The cost savings associated with a reduction in shielding thickness are expected to be more significant for the metal cask and concrete building designs because of their relatively higher costs. At the present time, it appears appropriate to use available designs from the vendors.

Based on the aforementioned vulnerabilities, the best shield would be the metal cask. Concrete shields are judged second best after metal based on their lack of dependence on any active systems. The water shield requires active systems for decay heat removal to prevent heatup and makeup to compensate for long term evaporation. It is also vulnerable to leaks from connected piping and its enclosing structure. Although water appears to be the least expensive shield material, its requirements for several active systems and qualified walls and floor actually make it one of the more expensive shields.

F.2.4 Structural Integrity

All of the spent nuclear fuel storage technologies are required to meet the same standards for structural integrity in accordance with appropriate codes. Structural integrity ensures that the confinement boundary around the spent nuclear fuel is maintained under all operational and accident conditions.

For incident-free operation, the dry storage designs are analyzed in terms of peak stresses on their canister and enclosing structure (i.e., metal cask, concrete cask, or vault). In wet storage designs, the fuel racks and pool structure are analyzed for operating loads. The source of these loads, in accordance with appropriate American Society of Mechanical Engineers codes, include such factors as deadweight, pressure, fill gas pressure, and thermal gradients.

For accident cases, additional loads are imposed upon the structures. These additional loads include seismic acceleration, high (or low) ambient temperature and solar heat flux, component drop or tip over, airflow passage blockage, external fire, tornado missile, flooding, etc. As with incident-free operation, specific prescribed margins of safety between the peak calculated stresses and the maximum allowable stress for a given component, location, and material must be maintained to substantiate structural integrity.

The principal structural-related differences between foreign research reactor spent nuclear fuel and commercial fuel for storage technology design purposes are:

- a typical foreign research reactor spent nuclear fuel element is much lighter [5 kg (11 lbs) as compared to 800 kg (1,760 lbs)] and shorter than a commercial fuel assembly (a stack of 5 typical foreign research reactor spent nuclear fuel elements is approximately equal in length to 1 commercial fuel assembly), and

- the strength of foreign research reactor spent nuclear fuel, in particular the predominant aluminum-clad design, is expected to be less than the commercial fuel assembly.

The much lower foreign research reactor spent nuclear fuel weight will reduce the total weight and load on the storage technology unit by about 19 metric tons (21 tons) for a 24-commercial fuel assembly design. For metal and concrete casks, this is a significant fraction of the total cask's weight, and can only improve the structural strength of the cask. The lower weight of the foreign research reactor spent nuclear fuel will increase the structural margins in the design and possibly allow for the use of less material in the structure compared with the commercial cask design. Any design changes to take advantage of the lower fuel weight would require detailed re-analysis, and are probably unnecessary.

The lower strength of the foreign research reactor spent nuclear fuel would require analyses to demonstrate that operational and postulated accident events do not result in structural failure of the fuel. However, since the principal means of confinement is the canister surrounding the fuel, its structural integrity is expected to be maintained, as it has already been qualified for the heavier commercial fuel under the same conditions and accidents.

Assuming that the same structural design limits apply for foreign research reactor spent nuclear fuel storage as for commercial fuel storage, the lower weight and strength of the foreign research reactor spent nuclear fuel would be expected to increase the original stress design margins.

The basket of any currently licensed cask would require redesign to accommodate the foreign research reactor spent nuclear fuel. Furthermore, it could be anticipated that permanently installed neutron poisons may be required in the basket to prevent criticality for the highly enriched fuels (initially 90 to 93 percent enrichment).

Each of the spent nuclear fuel storage technology designs that have been licensed by the NRC have undergone rigorous structural analyses and have been shown to meet all applicable standards and codes. Designs which have not yet been licensed would be required to present detailed structural analyses for review and confirmation to ensure structural integrity. No design has specific structural vulnerabilities that make it unsuitable for the storage of foreign research reactor spent nuclear fuel. It should be noted that any changes in existing NRC-approved storage designs that are deemed to impact stresses (i.e., reducing shielding wall thickness) would require extensive re-analysis and technical review for structural integrity. Thus, use of existing designs is favored.

F.2.5 Thermal Performance

Adequate decay heat removal is vital to preventing degradation of the fuel cladding barrier to fission product releases. The wet and dry storage technologies rely on a combination of conduction, convection (natural or forced), and radiation heat transfer mechanisms to ensure fuel cladding temperatures below appropriate long term storage limits.

In wet pool designs, fuel decay heat is transferred to the pool water by conduction and natural convection, which is induced by the axial enthalpy rise of the water as it passes over the active region of the fuel. An active cooling system consisting of redundant pumps, heat exchangers, and piping connected to the pool removes the heat in the bulk pool water. Careful thermal design of the spent nuclear fuel storage racks allows for sufficient natural convection flow over each fuel assembly to prevent any local nucleate boiling on the cladding surface throughout the pool. Therefore, the thermal performance of the pool technology relies on storage rack design for local thermal effects and an active external system for global heat removal. As previously discussed, this design has a long-established history of satisfactory performance. Wet storage can accommodate fuel of any power level.

The metal cask, dry storage design relies on a totally passive system for heat removal. The fuel decay heat, in an encapsulating inert gas atmosphere canister, is transferred to the canister's walls by a combination of radiation and conduction heat transfer. The canister walls, in contact with the metal (or sometimes metal sandwiched with a neutron-absorbing material) cask wall transfers this heat by conduction through the metal wall. At the outside of the metal cask, the heat is removed by conduction and natural convection to the environment. Some designs incorporate cooling metal fins on the exterior of the cask to enhance heat transmission to the air. The four metals used in spent nuclear fuel storage cask designs are ductile cast iron, carbon steel, lead, and stainless steel. In terms of their heat conduction properties, cast iron, lead, and carbon steel are superior to stainless steel because they have a thermal conductivity which is about three times that of stainless steel. The metal cask heat transfer system is not susceptible to thermal limits, since these metals have a higher temperature limit than fuel cladding. The only possible degradation of heat transfer could occur if the fuel canister seal was broken and the inert gas atmosphere lost. The sealing system is designed to withstand all postulated accidents and maintain integrity over the lifetime of the cask, because it constitutes part of the radioisotope confinement boundary.

As with metal casks, concrete casks use a passive heat removal system, but the concrete cask system has one inherent vulnerability. To remove fuel decay heat and stay below both the fuel cladding and concrete temperature limits, concrete casks must include a labyrinth airflow passage design that allows natural convection-driven air to enter the cavity enclosing the canister inside the concrete. The air then exits through higher elevation paths through the concrete to the environment. Concrete thermal conductivity is a factor of 10 to 40 lower composed of stainless steel or inconel, which have similar thermal conductivities to zircaloy, but a melting temperature of about 1,371°C (2,500°F). TRIGA fuel storage temperature limits are expected to be greater than for aluminum-clad fuel. The Savannah River Site is conducting a research and development project, initiated in FY 1994, to examine the applicability of aluminum-clad spent nuclear fuel dry storage.

At a minimum, a new thermal analysis would need to be performed for existing designs of spent nuclear fuel storage technologies. This analysis would use the parameters associated with foreign research reactor spent nuclear fuel instead of commercial spent nuclear fuel. The important changes in thermal performance parameters for the foreign research reactor spent nuclear fuel are:

- lower individual fuel assembly decay heat power,
- lower and/or different fuel temperature limits for aluminum-clad and stainless steel-inconel-clad fuels, and
- higher clad and fuel thermal conductivity for aluminum foreign research reactor spent nuclear fuel.

A temperature limit of 175°C (347°F) has been tentatively identified to avoid damage to the cladding of aluminum-clad spent nuclear fuel (Shedrow, 1994a and 1994b; Taylor et al., 1994). The results of this revised thermal analysis could impact the thermal design of the spent nuclear fuel storage technology. If the existing design results in unacceptable fuel and/or shielding temperatures, redesign could reduce the maximum heat load of each module or cask or increase the airflow passage area or height for concrete casks that rely on natural convection heat transfer. The new thermal analysis must take into account design restrictions that are imposed by criticality limits (i.e., the maximum allowable number of foreign research reactor spent nuclear fuel assemblies), and possible changes in shielding thickness due to lower gamma and neutron source terms that would improve the storage technology's thermal performance. Again, existing designs should be used to the greatest extent possible.

Commercial spent nuclear fuel dry storage systems require a minimum cooldown period of 5 years. For aluminum-clad foreign research reactor spent nuclear fuel, the preliminary cladding temperature limit of 175°C (347°F) becomes the determining criteria for dry storage loading above an average spent nuclear fuel element power level of 40 Watts each. Foreign research reactor spent nuclear fuel averages more than 40 Watts per element after a single year's discharge from the reactor and, if immediately placed into dry storage, would result in oversized facilities within several years as the radionuclides decay. Consequently, for the size of a foreign research reactor spent nuclear fuel dry storage facility to be minimized, an average foreign research reactor spent nuclear fuel power level below 40 Watts per element is necessary. On average, a 3-year cooldown period would be required. This results in the element's volume being the constraining criteria, and corresponds to maximum density of spent nuclear fuel (hence, minimum size of the facility) in the dry storage method. Consequently, the storage approach uses a minimum wet storage period of 3 years prior to emplacement into dry storage.

A comparative evaluation of the thermal performance of each fuel storage technology points to the metal cask and the solid concrete SILO as the simplest, effective, and least susceptible to any degradation. However, another design which has many merits is the concrete building. Although concrete buildings require open airflow passages to remove decay heat, size and a large elevation difference are factors which compensate for this weakness and make them good candidates. The concrete cask, with adequate design margins and surveillance is an acceptable thermal system. Finally, the wet pool system is a proven technology, but is dependent on an active system to remove heat. The inground concrete system in

Denmark (RISO National Laboratory) relies on a forced air active system and is characterized similar to the wet system in terms of its heat removal capabilities.

F.2.6 Ease of Use

For spent nuclear fuel storage, ease of use is defined as the lack of complexity involved in the process of loading spent nuclear fuel, and operating and maintaining the storage technology. For all storage designs, the spent nuclear fuel must be removed from the transportation cask to be placed into the storage facility, unless the design is a dual-purpose cask.

The technology that requires the fewest steps and lowest complexity for transferring spent nuclear fuel from the transportation cask to its storage location is wet pool storage. At a pool, the transportation cask is simply immersed under the water, opened underwater, and the fuel moved underwater to its final location in a storage rack in the pool. Pool water provides shielding, heat removal, and viewing of the fuel. The dry storage technologies all require additional intermediate steps, which include the insertion of fuel into a canister that must be subsequently drained of all water and air, seal welded, tested for leakage, and backfilled with inert gas. The canister is then placed into its dry storage structure (i.e., vault, concrete, or metal cask). The vault provides for this entire process within a shielded enclosing building, whereas the casks require transport by some vehicle between the transportation cask fuel transfer location and the cask site. Thus, for spent nuclear fuel transfer and loading, the wet storage design is easiest to use, followed by the dry vault.

After loading, operation of the storage facility is another important factor in determining ease of use. For operation, the individual metal or concrete casks are easiest, since they are designed as totally passive systems requiring only periodic visual inspection from a distance. The vault is slightly more complex than the casks because it includes a number of active systems (i.e., crane, power supplies, fuel handling machine) that may require some operational support. The wet storage is the most complex from an operational viewpoint because it includes a number of vital safety-related systems that must be monitored and controlled (e.g., heat removal system, water purification system, makeup water system, ventilation exhaust system).

Maintenance ease of use is closely related to operational ease of use since designs with more operational complexity require greater maintenance. Thus, the cask systems can be considered easiest, followed by the vault, and the wet technology.

In ranking the relative importance of the three aforementioned factors of ease of use (fuel loading, operation, and maintenance) the fraction of time spent during the life of the facility for fuel loading is expected to be much smaller than for operation and maintenance. Fuel loading will be a sporadic event over a long period of time, whereas operation and maintenance are considered continuous over this same period of time. Thus, operation and maintenance ease of use is given greater importance than fuel loading ease of use. With this ranking, the cask (both metal and concrete) technology is judged to have the greatest ease of use, followed by the vault [illegible] than dry metal casks, dry concrete casks, and dry concrete buildings. The dry concrete vault/building technology would be expected to have slightly higher operations and maintenance costs than the individual metal or concrete casks, since these buildings use active nonsafety systems such as lighting, cranes, and fuel drying dedicated to the vault facility.

For the construction of a new fuel storage facility for the purpose of storing foreign research reactor spent nuclear fuel on the order of approximately 23,000 assemblies, elements and/or rods, it is assumed that 5 "trimmed" foreign research reactor spent nuclear fuel assemblies would occupy the same approximate space as one commercial nuclear power plant fuel assembly (Boiling Water Reactor-type). "Trimmed" means that the non-essential portions (i.e., ends) of the spent nuclear fuel element have been removed, as detailed in Appendix B. Therefore, storage of the total amount of foreign research reactor spent nuclear fuel under consideration in this EIS would be the equivalent of about 5,000 commercial Boiling Water Reactor spent nuclear fuel assemblies. Since typical concrete, inground, or metal casks can store 52 power fuel assemblies, this foreign research reactor spent nuclear fuel inventory would require around 100 casks. A suitably sized single pool or concrete building could accommodate this inventory of spent nuclear fuel. Spent nuclear fuel storage manufacturers have indicated that metal casks typically cost about twice as much as concrete casks for the same quantity of fuel storage due to the higher costs of metal as compared to concrete. Based on its design, the least expensive concrete cask is expected to be the simple concrete SILO, since it does not have steel-lined internal air passages. The number of foreign research reactor spent nuclear fuel assemblies under consideration in this EIS may be amenable to the economic advantages that a single building or pool offers over a large number of individual casks. Another potential cost advantage of the pool or concrete building is that these are self-contained, not requiring access to any other facilities for the transfer of the fuel from the transport cask. Presented below is a brief summary of commercial cost experience with storage.

F.2.7.1 Costs for Dry Storage Designs

The cost for different spent nuclear fuel storage technologies varies significantly between designs. Some information on cost has been obtained from manufacturers and openly available literature. Relative order of magnitude cost data was obtained for the horizontal concrete NUHOMS module, vertical concrete Ventilated Storage Cask design, vertical concrete SILO, metal CASTOR vertical cask, and the modular dry vault concrete building design.

The principal elements of cost that should be considered for the storage of foreign research reactor spent nuclear fuel are: (1) engineering for redesign and licensing, (2) capital for the construction of the facility, and (3) operations and maintenance. In the interest of minimizing cost and schedule for the completion of any storage facility for foreign research reactor spent nuclear fuel, the licensing basis of 10 CFR 72 used by the NRC for commercial nuclear power plant spent nuclear fuel should be adopted for the foreign research reactor spent nuclear fuel. This regulation provides all the requirements for licensing foreign research reactor spent nuclear fuel storage and has been successfully applied to numerous dry spent nuclear fuel storage installations in the United States.

Redesign engineering should be limited to changes in the design of the basket that encapsulates the fuel, since foreign research reactor spent nuclear fuel has different dimensions, would probably be stacked, and could require different spacing and/or the incorporation of neutron absorbing plates to maintain subcriticality safety margins. Outside the basket, all remaining components should be identical to those already licensed for commercial nuclear fuel by the NRC, thereby significantly reducing engineering analysis and license review time and costs as well as drawing and specification changes. This could result in some overdesign in the shielding and heat removal of the system, but would have the benefit of greatly reduced engineering, licensing, and schedule costs. If thermal analyses show that unacceptable foreign research reactor spent nuclear fuel temperatures would occur in the storage facility, then more extensive redesign would be required (e.g., reduce excess concrete wall thickness not needed for shielding, which [illegible]

Information obtained on the unit capital costs for different storage designs shows a significant variation. The least expensive unit is the SILO due to its simple concrete-canister design and lack of internal air passage labyrinth. The most expensive unit cost, excluding the modular dry vault (which stores a larger number of fuel assemblies than the other storage designs), is the CASTOR metal cask, due to its use of a thick metal wall instead of concrete. The Ventilated Storage Cask and NUHOMS designs' costs fall between the SILO and CASTOR. If one were to rank, in decreasing order, the unit cost of the four cask designs, they would be: CASTOR, NUHOMS, Ventilated Storage Cask, and SILO. There is more than a factor of 10 difference between the SILO and the CASTOR.

An estimate of the capital costs for storing approximately 23,000 foreign research reactor spent nuclear fuel elements can be made with the following two assumptions. First, the average spent nuclear fuel assembly decay heat is between 10 and 40 Watts, which is reasonable and conservative based on the status of foreign research reactor spent nuclear fuel under consideration in this EIS. Second, five trimmed foreign research reactor spent nuclear fuel assemblies can be stacked to fit into the same approximate space as one commercial nuclear power plant spent nuclear fuel assembly (Boiling Water Reactor-type). Using these assumptions, 23,000 foreign research reactor spent nuclear fuel elements would require 375 SILOs, 100 VSC-24s, 100 NUHOMS-24Ps, or 150 CASTOR V21s. One sufficiently sized and designed modular dry vault would also accommodate the foreign research reactor spent nuclear fuel.

The operations and maintenance costs for all these designs are expected to be small based on utility

	Approximate Unit Capital Cost Range, $	*Approximate # of Canisters/Sleeves for Foreign Research Reactor Spent Nuclear Fuel*	*Total Capital Cost $M*	*Full-Time Equivalents, Loading/ Inspection*[b]	*Full-Time Equivalents Monitoring*[c]	*Other Annual Costs, $M*	*Total Annual Operating Cost $M*[d,e]
Metal Cask	800,000-1.1M	150 (max)	165	15 (max)	3 (max)	1	3.7
Horizontal Dry Storage Cask	400,000-500,000	100 (max)	50	15 (max)	3 (max)	1	3.7
Vertical Concrete Storage Cask	350,000	100 (max)	35	15 (max)	3 (max)	1	3.7
Modular Dry Vault	13,000/tube	5 foreign research reactor/vaul t tube	65	15 (max)	3 (max)	1	3.7
SILO	100,000	375 (max)	37.5	15 (max)	3 (max)	1	3.7

Reference for costs: (EPRI, 1993)

[a] *Intermediate wet pool required for dry storage facility not included because utilities already possess an on-site pool*

[b] *One shift operation*

[c] *Monitoring based on one Full-Time Equivalent per shift*

protection of the public from undue radiological risk. The provisions apply to all Departmental elements. This order references the storage of spent nuclear fuel. It also references instances where some DOE facilities are subject to provisions of 10 CFR 72. It was not made clear in the order which DOE facilities are subject to 10 CFR 72, which deals directly with all aspects of interim storage of spent nuclear fuel.

DOE Order 5480.22 (DOE, 1992b), entitled "Technical Safety Requirements," requires that DOE nuclear facilities delineate criteria, content, scope, documents, etc. The scope includes DOE elements, but excludes facilities exempt from NRC licensing and Naval Propulsion Program facilities. Although this order does not reference spent nuclear fuel, there are useful discussions of limiting conditions for operation of nonreactor nuclear facilities.

DOE Order 5633.3A (DOE, 1994e), entitled "Control and Accountability of Nuclear Materials," prescribes minimum requirements and procedures for control and accountability of nuclear materials at DOE facilities, which are exempt from NRC licensing requirements. By DOE definition, "nuclear materials" includes spent nuclear fuel. Storage of nuclear material is mentioned with respect to repositories.

DOE Order 6430.1A, (DOE, 1989a), entitled "General Design Criteria," has a section dealing with irradiated fissile material storage facilities (Section 1320). General criteria for nuclear criticality, confinement systems, effluent control and monitoring, and decontamination and decommissioning are discussed. Reference is made that, "the design professional shall consider the criteria provided in 10 CFR 72," (NRC, 1994) as well as NRC Regulatory Guides 3.49 (NRC, 1981) and 3.54 (NRC, 1984) for applicability to irradiated fissile material storage facilities. Other important standards for dry storage are ANSI/ANS-57.9 (ANSI, 1984a) and NRC Regulatory Guide 1.13 (NRC, 1975).

F.2.9 Aluminum-Clad Research Reactor Spent Nuclear Fuel Dry Storage Experience

F.2.9.1 Australia

Australia has successfully operated an underground dry storage facility for High Flux Australian Reactor MTR-type aluminum-clad research reactor spent nuclear fuel for 31 years at the Lucas Heights Facility (Australia, 1993; Ridal, 1994; Silver, 1993). The facility consists of a building enclosing a concrete floor with 50 steel plugs that are bolted to a steel collar set into the concrete.

Each plug covers a stainless steel-lined 0.64 cm (0.25 in) thick, 14 cm (5.5 in) inner diameter, 15.2 m (50 ft) deep borehole tube that is sealed at the bottom. The rock around these 50 borehole tubes is sandstone with a variable clay matrix and bands of enriched siderite. The actual boreholes in the sandstone are 16.5 cm (6.5 in) in diameter, 16.8 m (55 ft) deep, and spaced 1.14 m (45 in) center-to-center apart.

Each borehole liner is filled with 11 stainless steel canisters that hold 2 stacked fuel assemblies each. The borehole liner is evacuated and backfilled with dry nitrogen. The borehole liner plug is designed with its own plug to allow for atmosphere purging, backfilling, and annual monitoring of any fission product gases that would indicate canister breach.

The stored High Flux Australian Reactor spent nuclear fuel is uranium-aluminum alloy with aluminum cladding in the shape of four concentric tubes. Each fuel assembly has an outer diameter of 10 cm (3.93 in) and a length of 66 cm (26 in). The ^{235}U content for each fresh fuel assembly was 170 g (0.37 lbs), and the ^{235}U was enriched to 60 percent. Fuel has been stored at this facility for 8 to 31 years with no radioactivity releases or evidence of corrosion over this time period. No nuclear poisons for criticality safety or heat transfer analyses were deemed necessary because of the relatively low ^{235}U content, large borehole spacing, and low fuel assembly decay heat. The storage criteria for each fuel assembly is a maximum decay heat of less than 4.5 Watts, which after 20 years, drops to 1.5 Watts per fuel assembly. Fuel examined in a hot cell after 10 and 25 years of storage at this facility showed no visible signs of corrosion. Figure F-29 illustrates the facility design.

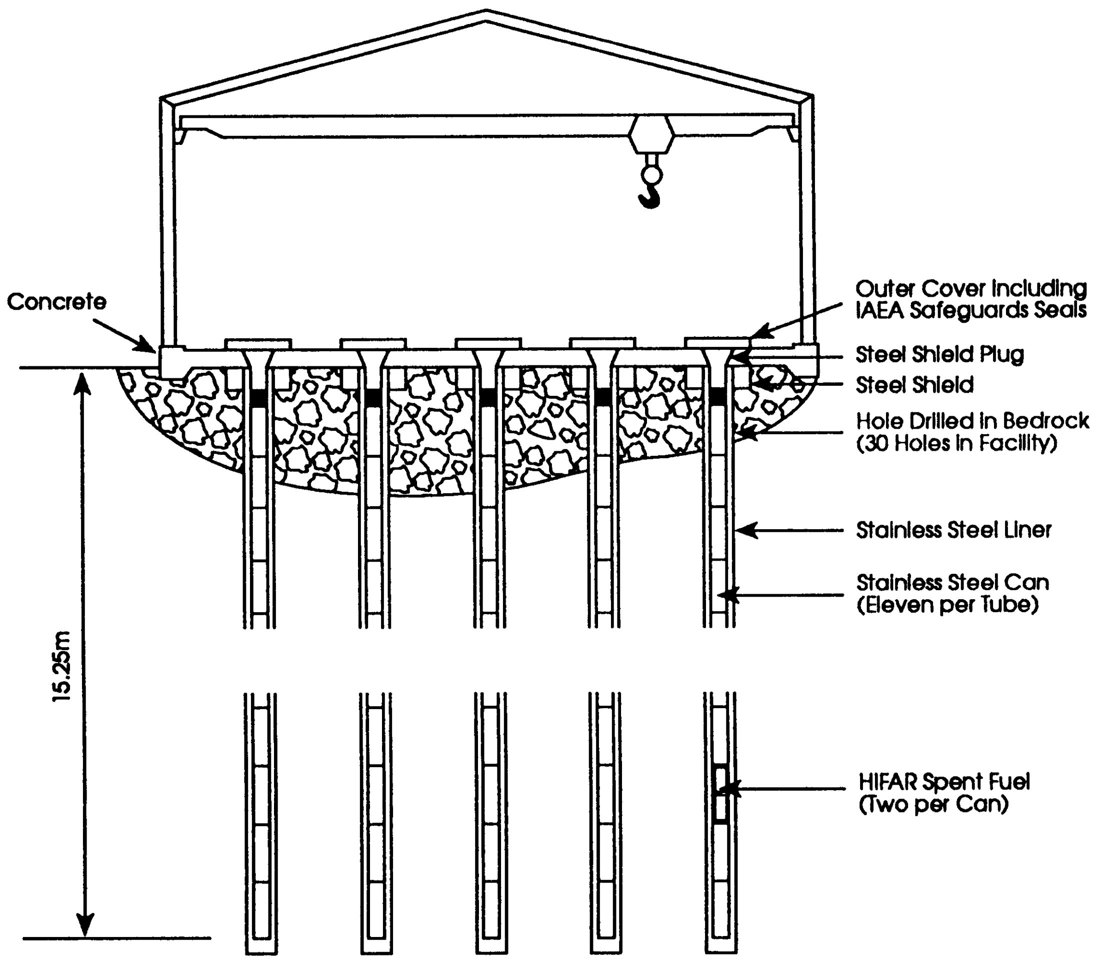


Figure F-29 High Flux Australian Reactor Spent Nuclear Fuel Dry Storage Facility

F.2.9.2 Japan

In 1982, The Japan Atomic Energy Research Institute completed construction of a dry spent nuclear fuel storage facility at Tokai, Japan for the storage of JRR-3 research reactor spent nuclear fuel (Shirai et al., 1991). The facility consists of a building enclosing several support areas (cask receipt, loading, cask maintenance, and control room) and the drywell storage structure (Figure F-30). The storage structure is 12 m (39.4 ft) long, 13 m (42.7 ft) wide, 5 m (16.4 ft) high concrete box that encapsulates a 10 x 10 lattice array of drywells (Figure F-31). Each drywell storage canister (Figure F-32) comprises a

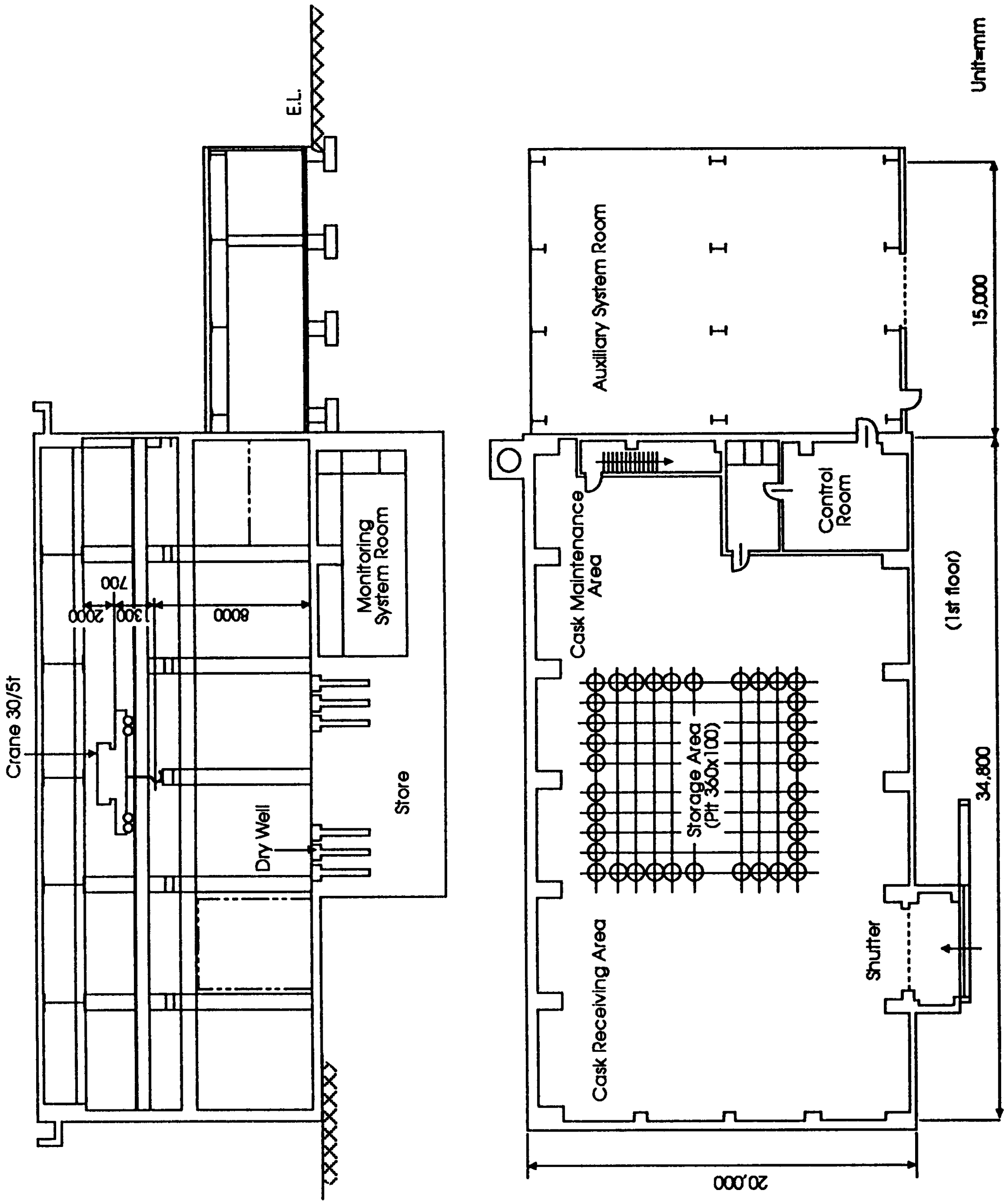


Figure F-30 General Arrangement of Dry Storage Facility at Tokai, Japan

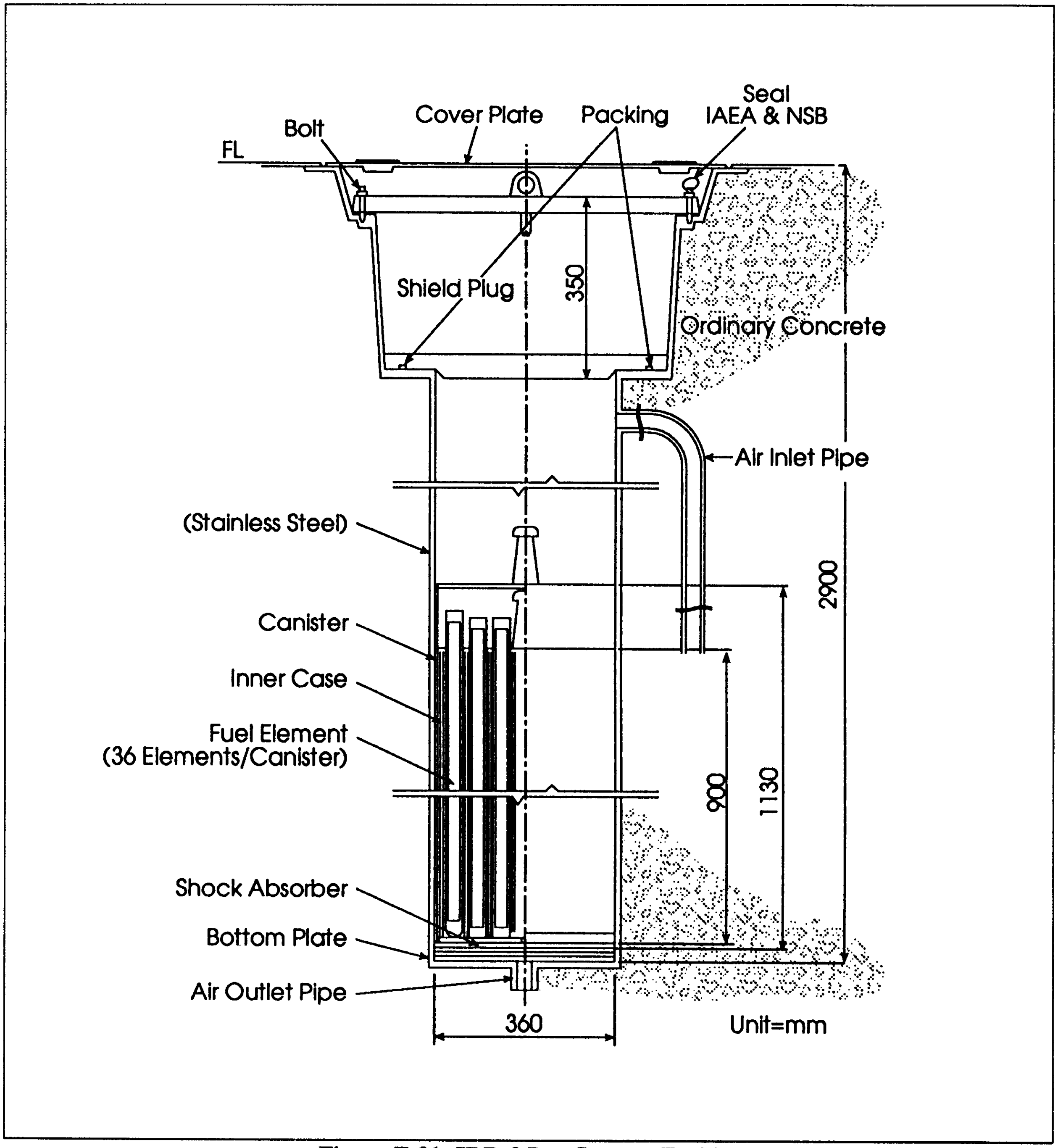


Figure F-31 JRR-3 Dry Storage Facility

0.8 cm (0.3 in) thick stainless steel liner 2.5 m (8.2 ft) deep and has a 36 cm (14.2 in) inner diameter. Each drywell has a labyrinth air inlet and outlet pipe for radiation monitoring and decay heat removal, and is

covered with a 35 cm (13.8 in) thick carbon steel shield plug. The plug is bolted to the concrete

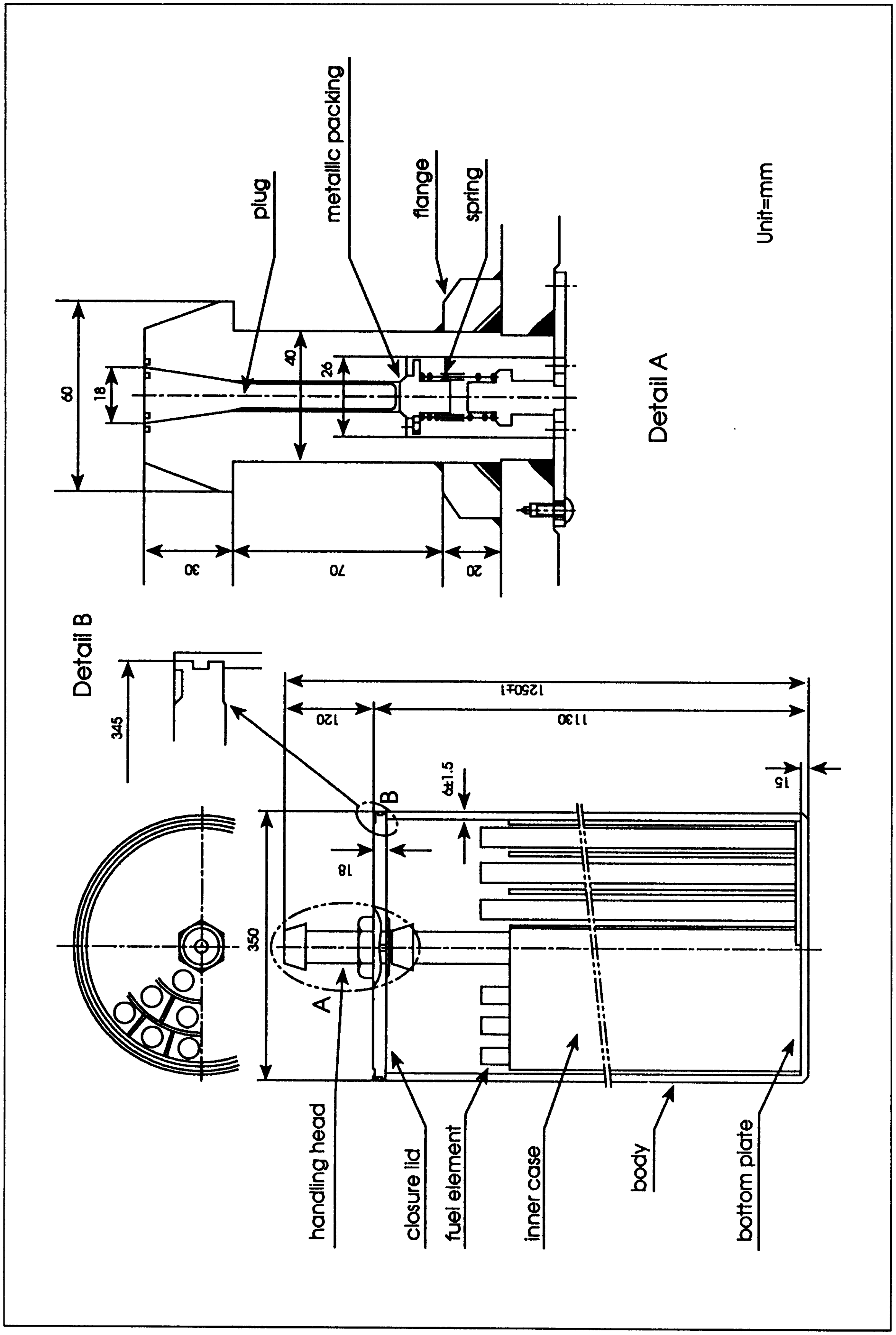


Figure F-32 Storage Canister

in aluminum cladding. The element is about 95 cm (37.4 in) long, with a 2.5 cm (1 in) outer diameter. Before storage, each element must meet the specifications of:

- maximum burnup = 800 Megawatt Days per metric ton,
- minimum cooling time = 2,500 days (6.8 years),
- maximum fission product activity = 110 Ci, and
- maximum decay heat = 0.5 Watt.

Although the low decay heat for the fuel elements eliminates the need for any cooling system (i.e., natural convection and conduction are sufficient), a system of blowers, filters, dehumidifiers, and monitors is provided for the facility. This system is used to provide subatmospheric pressure in the drywell so that any

minimizing any corrosion, and is also part of the radiation monitoring system designed to sample airflow for any escaping Krypton-85, a long-lived fission product that would be indicative of canister and fuel degradation. Maximum fuel storage temperature is maintained below 45°C (113°F) in this storage facility.

three approaches are estimated to require less than 4.5 ha (11 acres) of site land for receipt of all of the foreign research reactor spent nuclear fuel under consideration in this EIS.

F.3.1 Dry Storage Facility Designs

F.3.1.1 Spent Nuclear Fuel Storage Using a Dry Vault (Modular Dry Vault Storage)

As noted previously, the dry vault facility is an aboveground, self-contained concrete structure that includes dry fuel loading and unloading (Fort St. Vrain, 1992; Shedrow, 1994a and 1994b; Taylor et al., 1994; Claxton et al., 1993). The vault approach design consists of four major components: a receiving/loading area, fuel storage canisters, a shielded container handling machine, and a modular array for storing the fuel storage canisters (Figure F-33). The receiving area uses a small wet pool for unloading the transportation casks and for short-term (1 to 3 year) storage of foreign research reactor spent nuclear fuel exceeding 40 Watts per element. Table F-17 summarizes some typical modular dry vault storage parameters. The vault consists of several array units, and each unit provides storage for hundreds of fuel elements. The vault itself consists of a charge/discharge bay with a fuel handling machine above a floor containing steel tubes that house the (removable) fuel canisters. Shielding above the spent nuclear fuel is provided by the thick concrete floor and shield plugs inserted into the top of the steel storage tubes. The steel tubes serve as secondary containment for the foreign research reactor spent nuclear fuel and descend into an open storage area. Large labyrinth air supply ducts and discharge chimneys permit natural convection cooling of the steel spent nuclear fuel storage tubes, while the perimeter concrete walls provide for shielding. The design allows for expansion by adding additional units of arrays to the end of the vault, or by construction of another module. The vault facility also includes a receiving and loading bay that allows handling of the shielded transportation casks and unloading of the foreign research reactor spent

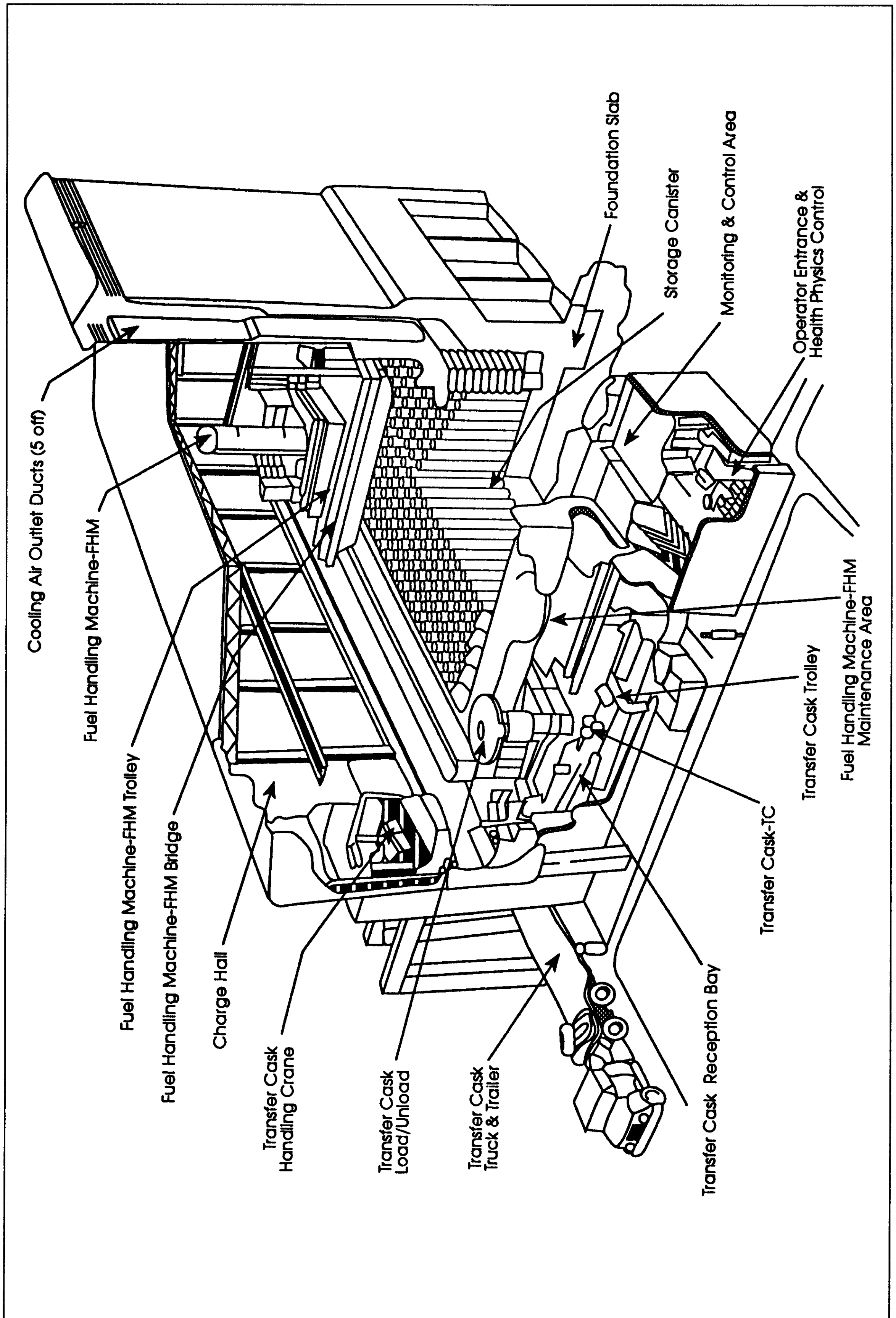


Figure F-33 Vault Elevation View

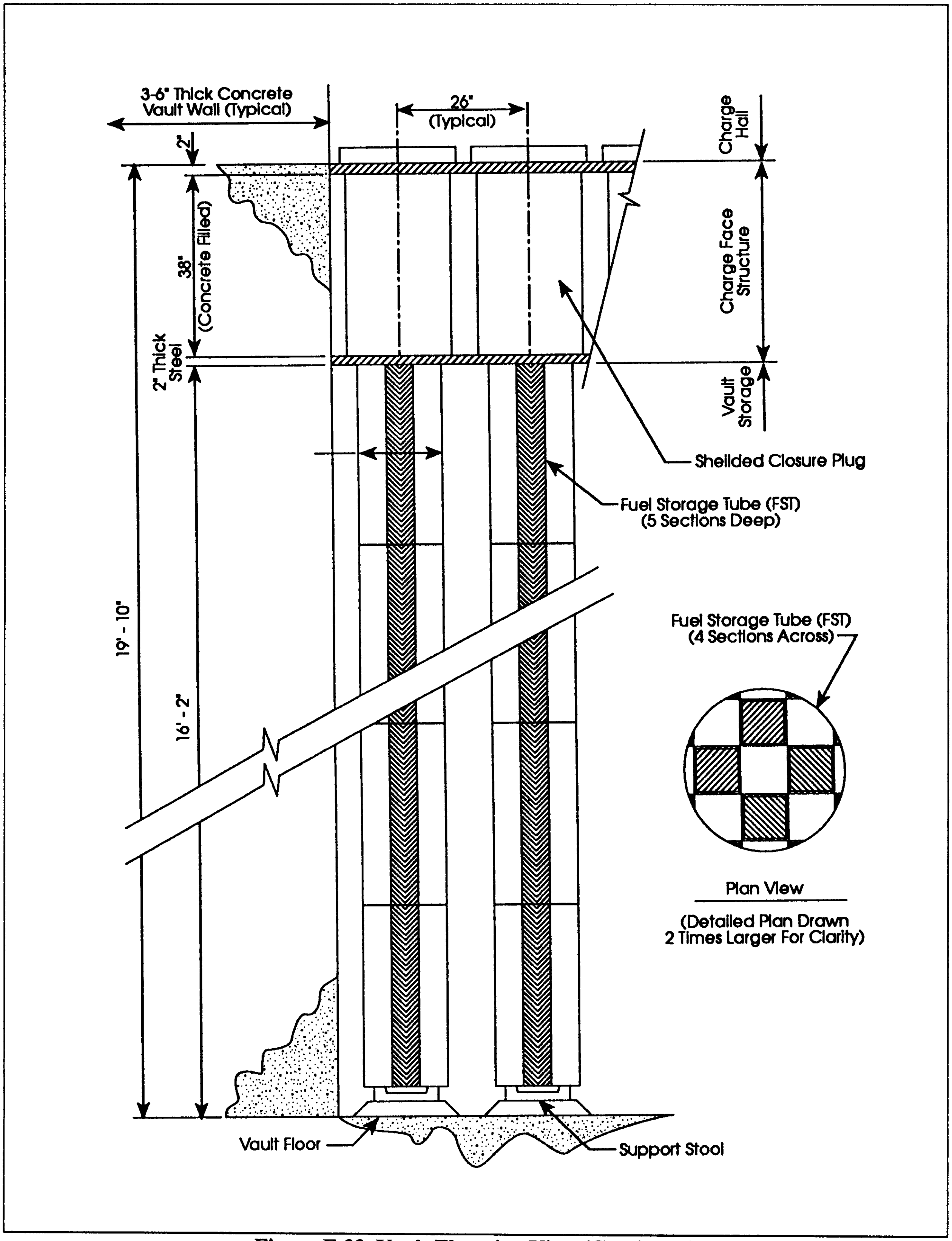


Figure F-33 Vault Elevation View (Continued)

Table F-17 Summary of Modular Dry Vault Storage Parameters for Foreign Research Reactor Spent Nuclear Fuel[a]

Construction Phase:	
Disturbed Land Area	3.7 ha (9 acres)
Facility: Size (Area) Concrete Steel	 5,000 m^2 (54,000 ft^2) 21,800 m^3 (28,500 yd^3) 5,200 mt (5,750 tons)
Soil Moved	11,000 m^3 (14,400 yd^3)
Equipment Fuel	835,000 l (221,000 gal)
Construction Debris/Waste	1,800 m^3 (2,400 yd^3)
Work Force	190/yr average, 234/yr peak
Duration (Years)	4 years for construction, 1.5 years for design
Capital Cost	$370 million[b]
Operation Phase:	
Electricity	800 - 1,000 MW-hr/yr (staging facility)
Water	2.1 million l/yr (550,000 gal/year) for first 10 years, 0.9 million l/yr (238,000 gal/yr) thereafter
Wastestreams	
Solid Low Level Waste	22 m^3/yr (780 ft^3/yr) during receipt, 1 m^3/yr (35 ft^3/yr) thereafter
Waste Water	1.59 million l/yr (420,000 gal/yr) during receipt, 0.4 million l/yr (109,000 gal/yr) thereafter
Staff (Full-Time Equivalents)	30 during receipt, 8 thereafter
Annual Operating Cost	$15.6 million during handling, $0.6 million during storage[b]

[a] *Staging facility parameters are based upon the regionalized, small wet pool (Dahlke et al., 1994).*

[b] *Cost estimates are in $1993 (EG&G, 1993)*

- spent nuclear fuel decay heat between 40 and 80 Watts per element: 32 vault units; and
- spent nuclear fuel decay heat between 10 and 40 Watts per element: 28 vault units.

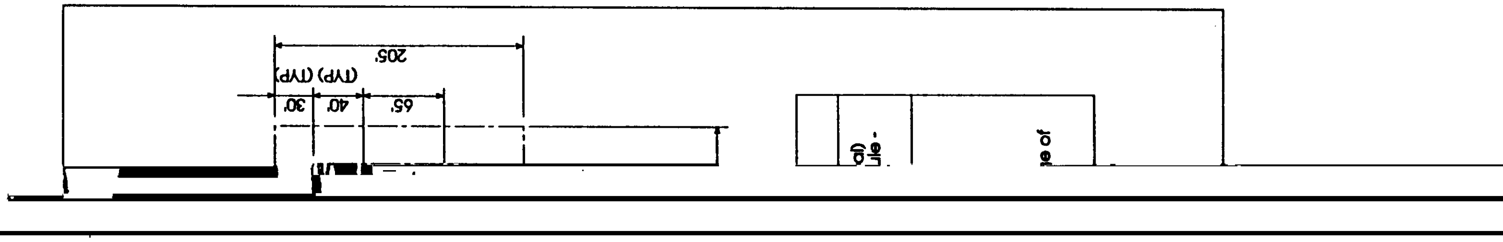
For "cold" fuel (10 Watts per element), potentially more than 44 spent nuclear fuel canisters could be placed per vault unit. This would require a customized design. Figure F-34 displays the 10 to 40 Watts and 80 Watts per element cases.

Higher decay heat foreign research reactor spent nuclear fuel would have to be temporarily stored in the small wet storage pool. The storage period is not expected to exceed 3 years.

Criticality concerns are addressed by fuel geometry within the canister and by the use of nuclear poisons (e.g., borated steel in the baskets, etc.). Vault geometry is used to maintain a minimum spacing between adjacent fuel elements or groups of fuel elements to prevent criticality. Nuclear poisons absorb neutrons, thus preventing criticality.

The vault/canyon design has been licensed by the NRC for a specific site. It represents a complete stand-alone facility that can be dedicated to foreign research reactor spent nuclear fuel without requiring the utilization of any other facilities at the host site. Cask handling, maintenance, spent nuclear fuel loading, spent nuclear fuel inspection, spent nuclear fuel storage, and (potentially) characterization can all be accomplished within the same facility.

The cost to construct a modular dry vault storage facility with a staging area sufficient to unload, characterize, can, temporarily store in a small pool, and transfer the spent nuclear fuel to the vault storage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area is estimated to be $370 million. The annual operating cost for this facility is estimated to be $15.6 million during the period of handling and transfers of the spent nuclear fuel and $0.6 million during the period of storage. The cost estimate for the facility is based on a cost report prepared by Idaho Inc. (EG&G, 1993) with the addition of the cost of a small wet storage facility reported by Dahlke et al. (Dahlke et al., 1994).

F.3.1.2 Spent Nuclear Fuel Storage Using Dry Casks

The dry cask storage approach consists of the following components (BG&E, 1989; Duke Power Company, 1988; Shedrow, 1994a and 1994b; Taylor et al., 1994; Claxton et al., 1993):

- a staging facility for cask receipt and unloading and for loading foreign research reactor spent nuclear fuel into the dry storage casks [a wet pool is used for this purpose, and for short-term (1 to 3 years) storage of foreign research reactor spent nuclear fuel with a heat load exceeding 40 Watts per element],

- an inspection/characterization facility, for examining fuel integrity and canning degraded spent nuclear fuel as required (this may be incorporated into the staging facility as an inspection cell or be immediately adjacent to it),

- a dry storage cask (usually concrete) [this provides for the shielding and the structural stability of the spent nuclear fuel storage],

- a transfer mechanism, such as a dedicated truck/trailer combination with a ram for horizontal modules, or a crane for vertical modules, and

- a separate fuel canister which may or may not be used [if used, it is typically around 4.6 m (15 ft) long and 1.7 m (5.5 ft) in diameter and weighs around 32 metric tons (36 tons)].

The dry cask approach requires the staging facility to receive and inspect the spent nuclear fuel shipment. The transportation cask would be unloaded in a small wet pool within the facility. Subsequently, the spent nuclear fuel is loaded into the dry cask (or spent nuclear fuel canister for the horizontal cask), and the cask is placed upon an outside concrete slab. The horizontal approach uses a dry spent nuclear fuel transfer canister for containing the spent nuclear fuel. This is placed within a shielded transfer cask and moved to the outside modular storage facility. A hydraulic ram inserts the transfer canister inside the horizontal storage module, followed by sealing with a shield plug.

The dry storage modules are designed to withstand normal loads and design basis accident effects, such as earthquakes, tornadoes, and floods. The concrete provides radiation shielding for gamma rays and neutrons. Natural air circulation dissipates the heat as air enters through inlet vents near the bottom of the cask, passes around the spent nuclear fuel canister, and exits near the top. Screens and grills keep birds and other animals out of the cooling duct area. Some of the candidate sites have facilities which may be used for cask receipt and unloading and spent nuclear fuel inspection and transfer to storage.

The application of dry cask storage technology to foreign research reactor spent nuclear fuel depends upon the heat load. Horizontal casks are anticipated to be slightly more restrictive than the vertical casks with respect to the heat load, and are thus the focus of the discussion. The standard design for a horizontal fuel canister provides for 24 or 52 sleeves (i.e., Pressurized Water Reactor or Boiling Water Reactor spent nuclear fuel, respectively), each about 4.6 m (15 ft) long. As with the vault approach, it is conservatively assumed that each sleeve contains five foreign research reactor spent nuclear fuel elements (i.e., in layers), within a basket or can arrangement for maintaining spacings and retrievability. As with the vault approach, the number of dry storage casks depends upon the decay heat of the spent nuclear fuel and a cladding temperature limit [175°C (347°F) for aluminum cladding, with an air inlet temperature of 49°C (120°F)]. The 24-sleeve design allows for a maximum of 120 elements of foreign research reactor spent nuclear fuel with 40 to 80 Watts per element of decay heat, while the 52-sleeve design provides for a maximum of 260 elements per dry storage cask. Thus, the number of casks required is as follows:

- decay heat between 40 and 80 Watts per element: 205 casks, and
- decay heat between 10 and 40 Watts per element: 94 casks.

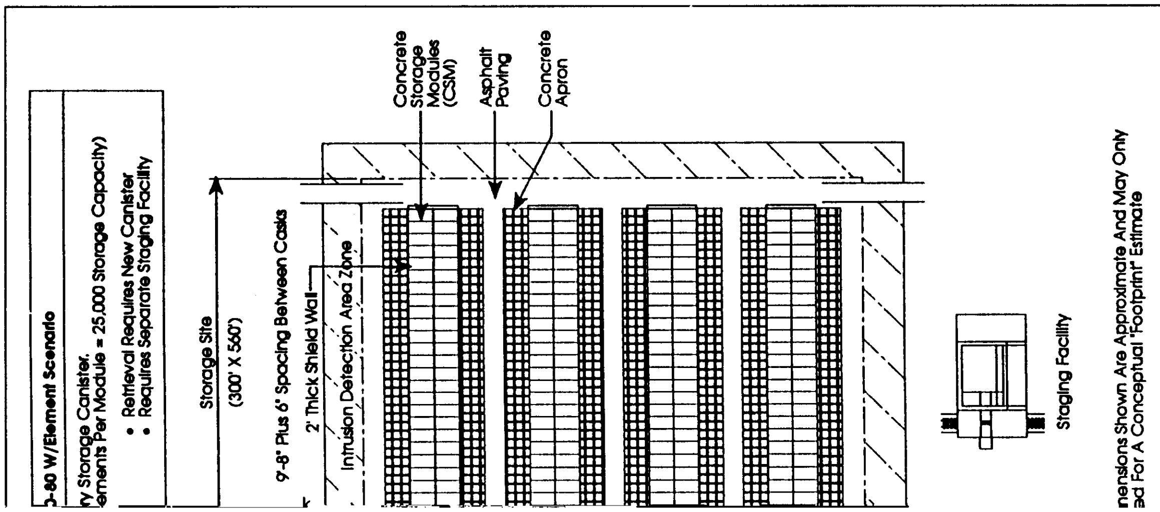
Note that these values are very conservative and correspond to a maximum of around 40 percent of the NRC-licensed heat loads per cask. Initially, foreign research reactor spent nuclear fuel with higher heat loads could be unsuitable for the dry storage cask pending detailed heat transfer analysis and a determination of limiting fuel storage temperature for aluminum-clad and TRIGA spent nuclear fuel. However, such relatively high decay heat fuel represents a small percentage of the currently identified foreign research reactor spent nuclear fuel so that its impact would be small; and after 1 to 5 years of wet storage, it would all be below a heat duty of 80 Watts per elements. The storage approach assures a minimum spent nuclear fuel wet storage time of 3 years after discharge prior to dry storage. This would essentially ensure that all foreign research reactor spent nuclear fuel is below a heat output of 40 Watts per assembly.

Figure F-35 displays approximate layouts for the dry cask storage facility predicated upon a horizontal cask design. Table F-18 summarizes some general parameters of dry cask storage.

The dry storage cask technology requires a separate staging facility for foreign research reactor spent nuclear fuel unloading, canning, and storage cask loading and transportation cask maintenance. This facility has the following operational areas:

- *Transportation Cask Handling*: This incorporates cask maintenance, truck/railcar unloading, decontamination/washdown, radioactive material control, and cask sampling/flushing/degassing.
- A small wet pool for fuel transfer and short-term storage.
- *Spent Nuclear Fuel Unit Handling*: Fuel removal, decontamination, fuel drying, fuel canning, inerting, and thermal measurements.
- *Spent Nuclear Fuel Unit Transfer*: This constitutes placement of the spent nuclear fuel into the cask or canister, followed by sealing.
- *Radwaste Treatment*: This includes collection, treatment, and preparation for disposal of contaminated effluents and radwaste treatment and solidification.
- *Heating, Ventilation, and Air Conditioning*: This represents heating, ventilation, and air conditioning of the facility so that contamination of the workers and the environment is avoided.

The inspection/characterization facility includes a shielded dry hot cell for spent nuclear fuel analysis and examination, and canning of degraded spent nuclear fuel. All equipment and instrumentation within the cells is remotely operated to provide chemical, physical, and radiological properties, as needed. The facility is maintained under negative pressure with exhaust through High Efficiency Particulate Air filters



atts Scenario and 40 to

Table F-18 Summary of Dry Cask Storage Parameters for Foreign Research Reactor Spent Nuclear Fuel[a]

Construction Phase:	
Disturbed Land Area	3 ha (7.7 acres)
Facility:	
Size (Area)	2,200 m^2 (24,000 ft^2)
Concrete	17,500 m^3 (22,900 yd^3)
Steel	4,500 metric tons (5,000 tons)
Soil Moved	11,000 m^3 (14,400 yd^3)
Equipment Fuel	810,000 l (214,000 gal)
Construction Debris/Waste	1,800 m^3 (2,400 yd^3)
Work Force	50/yr for staging facility, 50 per 25 cask array, 1 array/yr
Duration (Years)	4 years for construction, 1.5 years for design
Capital Cost	$366 million[b]
Operation Phase:	
Electricity	800 - 1,000 MW-hr/yr (staging facility)
Water	2.1 million l/yr (550,000 gal/yr) during receipt, 0.9 million l/yr (238,000 gal/yr) thereafter
Wastestreams	
Solid Low Level Waste	16 m^3/year (565 ft^3/year) during receipt, 1 m^3/yr (35 ft^3/yr) thereafter
Waste Water	1.58 million l/yr (412,000 gal/year) during receipt, 0.4 million l/yr (109,000 gal/yr) thereafter
Staff (Full-Time Equivalents)	30 during receipt, 8 thereafter
Annual Operating Cost	$17.3 million during handling, $0.3 million during storage[b]

[a] *Staging facility parameters based upon the Regionalized, Small Wet Pool (Dahlke et al., 1994)*

[b] *Cost estimates are in $1993 (EG&G, 1993)*

to mitigate the environmental effects of any radionuclide releases. This facility is normally located immediately adjacent to, or within, the staging facility.

Dry cask storage is unique among the three storage technologies because of its ability to be operationally integrated with existing facilities, which allows for faster implementation as compared to the other two storage technologies. Several DOE sites have facilities with spent nuclear fuel handling capabilities similar to the requirements of the staging facility. Potential examples include the RBOF at the Savannah River Site and the ICPP-666 storage pool area. For dry cask storage, the spent nuclear fuel would be shipped to the existing facility and unloaded from the transportation cask. The spent nuclear fuel would be inspected, canned if identified as a degraded element, and placed inside the storage canister. Spent nuclear fuel with heat loads exceeding 40 Watts per element would be stored in the existing facility to allow cooldown prior to cask storage. After filling, the canister would be sealed and placed inside the storage cask. The only new construction required would be the concrete storage pad (for vertical casks) or the concrete storage modules (for horizontal casks). For the foreign research reactor spent nuclear fuel receipt rate of approximately 2,000 elements per year considered in the analyses in this EIS, approximately 8 storage casks would be needed annually.

The cost to construct a dry cask storage facility with a staging area sufficient to unload, characterize, can, temporarily store in a small pool, and transfer the spent nuclear fuel to the cask storage area is estimated to be $366 million. The annual operating cost for this facility is estimated to be $17.3 million during the period of handling and transfers of the spent nuclear fuel and $0.3 million during the period of storage.

The cost estimate for the facility is based on a cost report prepared by Idaho Inc. (EG&G, 1993) with the addition of the cost of a small wet storage facility reported by Dahlke et al. (Dahlke et al., 1994).

F.3.2 Wet Storage Facility

Three generic wet storage facility options have been proposed for foreign research reactor spent nuclear fuel. They are denoted Centralized-Underwater Fuel Storage Facility, Regionalized Large-Underwater Fuel Storage Facility, and Regionalized Small-Underwater Fuel Storage Facility (Dahlke et al., 1994). The difference between these 3 options is that Centralized-Underwater Fuel Storage Facility is sized to store 100 percent of the foreign research reactor spent nuclear fuel under consideration in this EIS (Figure F-36). Regionalized Large-Underwater Fuel Storage Facility is designed for the storage of

75 percent of the foreign research reactor spent nuclear fuel, and Regionalized Small-Underwater Fuel Storage Facility will accommodate 25 percent of the foreign research reactor spent nuclear fuel. These three options were selected to encompass any conceivable decision regarding centralization or regionalization (by geography or fuel type) for the foreign research reactor spent nuclear fuel storage sites. The design features of all three wet storage facility options are identical with the exception that building and pool sizes and, in the case of the Regionalized Small-Underwater Fuel Storage Facility, the number of storage pools and receiving bays is smaller for the Regionalized Large-Underwater Fuel Storage Facility and Regionalized Small-Underwater Fuel Storage Facility. Table F-19 presents the difference in design between these three facilities. Because the design and environmental impacts of the larger Centralized-Underwater Fuel Storage Facility would bound the two smaller facility designs, the balance of the presentation in this section addresses the specific design of the Centralized-Underwater Fuel Storage Facility for storage of 100 percent of the foreign research reactor spent nuclear fuel.

The proposed new wet storage facilities consist of a fuel storage area and support areas (Dahlke et al., 1994). The Fuel Storage Area provides for the receipt of cask transportation vehicles, cask unloading and decontamination, fuel handling, transfer, and storage. Support areas provide for the equipment necessary to maintain and operate the storage area (e.g., heating, ventilation, air conditioning, water treatment, and waste management). The wet storage facility would be constructed as a structure that meets all current nuclear regulations for withstanding natural events such as seismic, tornado, and flood, as well as aircraft impact loads. All systems supporting the operation of the fuel storage facility would also meet these safety requirements. The facility is equipped with a 118-metric ton (130-ton) overhead cask handling crane, and a 9-metric ton (10 ton) fuel handling crane. Each cask transportation vehicle would enter the facility through one of two bays, where it would be monitored and washed from transportation dust. When the external surfaces are cleaned, the cask would be placed into a decontamination room where the cask would be prepared as needed to facilitate underwater unloading. The cask would then be placed in an unloading pool. Transportation casks would be monitored and, if clean of radioactive contamination, placed in an unloading pool. The cask receiving area can accept two simultaneous shipments on 3 m (10 ft) by 24.4 m (80 ft) trucks or railcars and casks weighing up to 114 metric tons (126 tons), each with a total individual cask and transport vehicle weight of 177 metric tons (195 tons).

There are two stainless steel-lined unloading pools, one measuring 6.4 m (21 ft) long, 5.8 m (19 ft) wide by 13.4 m (44 ft) deep, and the other measuring 6.1 m (20 ft) long, 6.1 m (20 ft) wide, and 11 m (36 ft) deep. There are two decontamination hot cells. Each unloading pool has a cask washdown system. Prior to being placed in one of the two storage pools, each fuel element would be checked to ensure that it is properly configured for direct transfer to the fuel storage pool buckets. If not, it would be transferred to the fuel cutting/canning pool, which is 10.4 m (34 ft) long, 5.8 m (19 ft) wide, and 9.4 m (31 ft) deep. Here it would be prepared for transfer to the storage pool buckets.

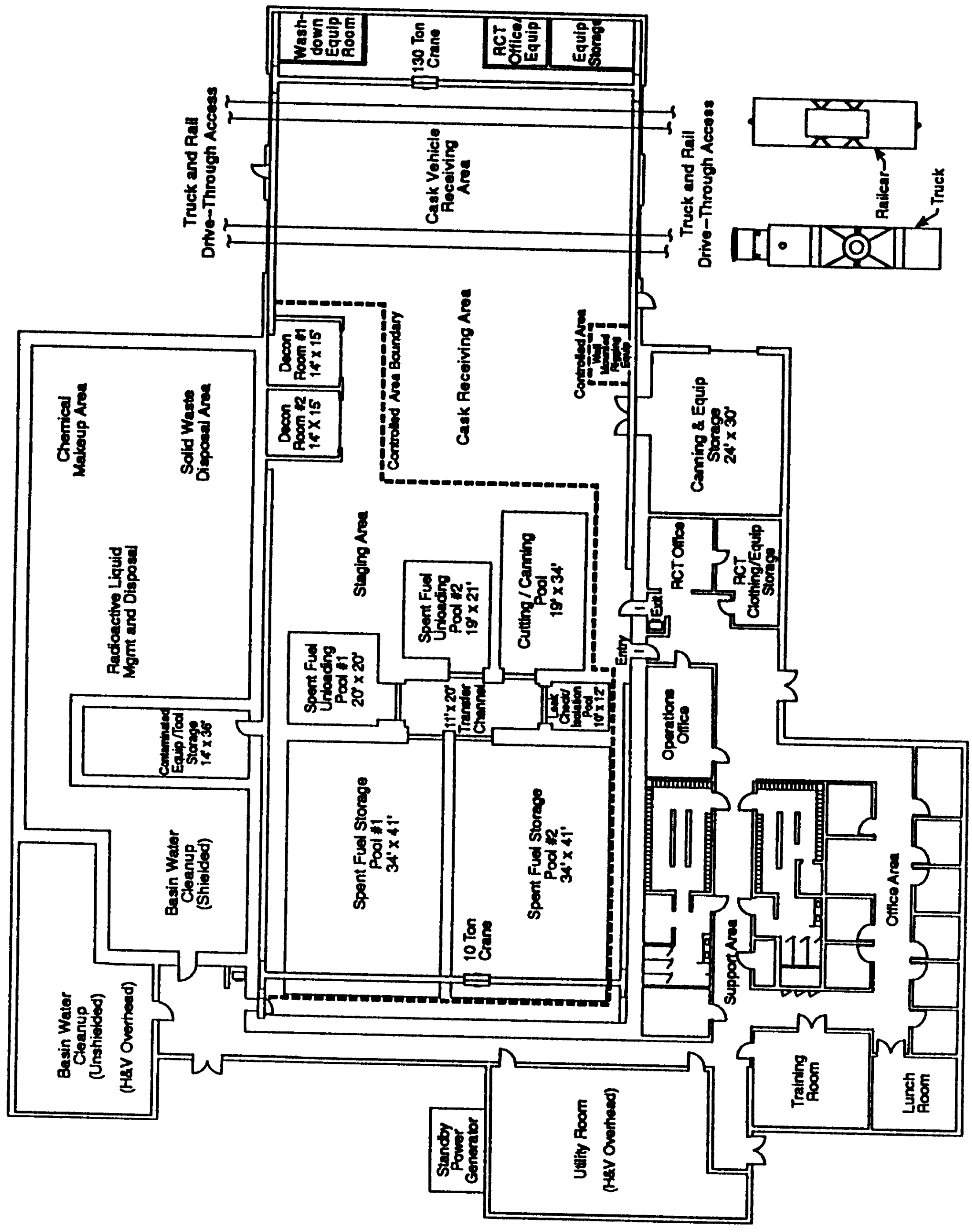
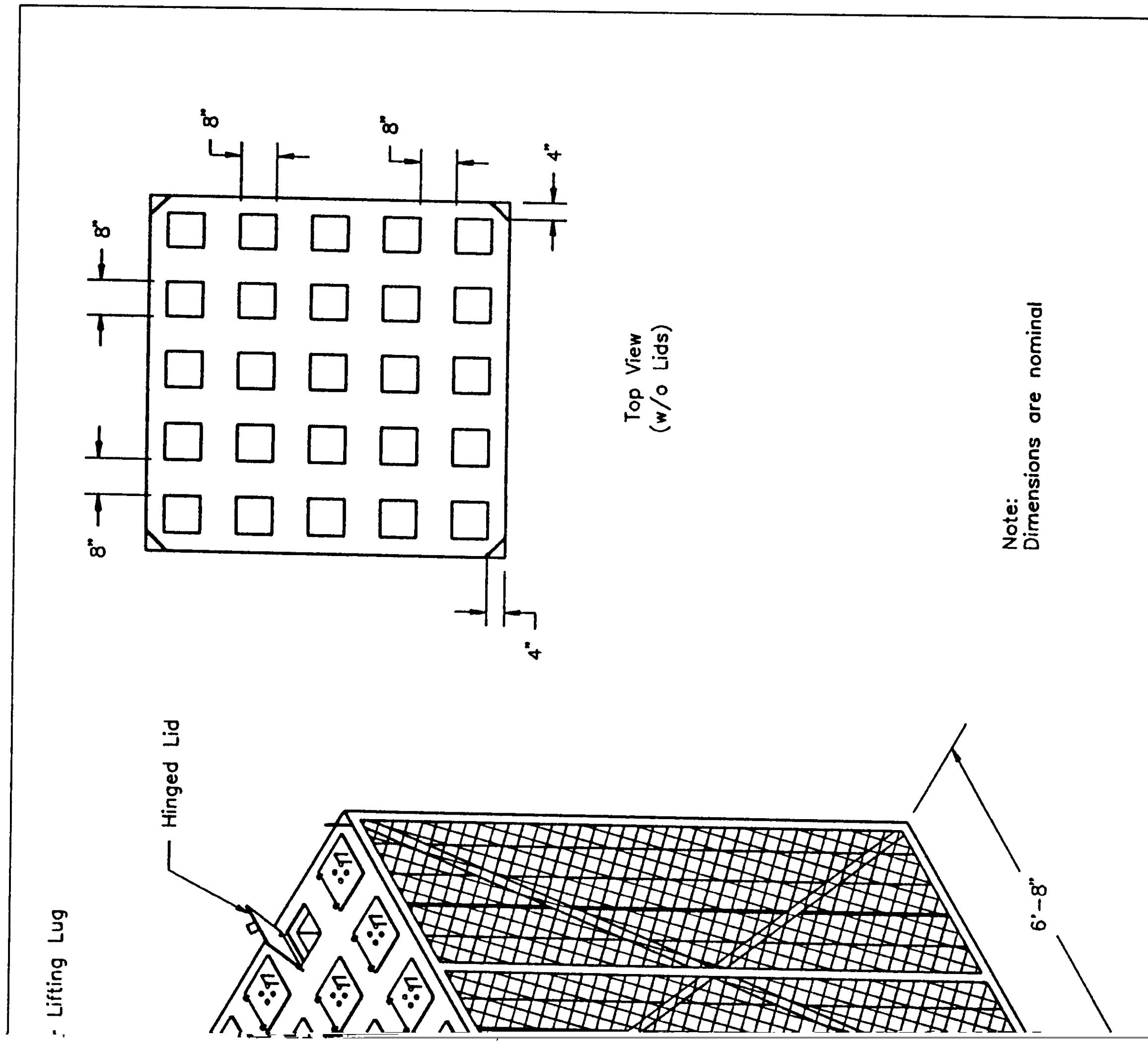


Figure F-36 Generic Wet Storage Facility for All of the Foreign Research Reactor Spent Nuclear Fuel

Table F-19 Design Difference Between 100 Percent, 75 Percent, and 25 Percent Generic Wet Storage Facilities

Design Parameter	*Wet Storage Capacity (Amount of Foreign Research Reactor Spent Nuclear Fuel)*		
	100%	*75%*	*25%*
Number of Storage Pools	2	2	1
Storage Pool Length and Width, m (ft)	16.5 x 10.4 (54 x 34)	12.5 x 10.4 (41 x 34)	10.4 x 8.2 (34 x 27)
Transfer Channel Length and Width, m (ft)	6.1 x 3.4 (20 x 11)	6.1 x 3.4 (20 x 11)	6.1 x 3 (20 x 10)
Fuel Unloading Pool Length and Width, m (ft)	6.4 x 5.8 (21 x 19)	6.4 x 5.8 (21 x 19)	6.1 x 6.1 (20 x 20)
Number of Receiving Bays	2	2	1

If cask measurements indicate that fuel is degraded, the fuel would be transferred to the isolation pool which is 3.7 m (12 ft) long, 3 m (10 ft) wide, and 9.4 m (31 ft) deep. This pool is equipped so that wet [illegible]



gure F-37 Storage Racks

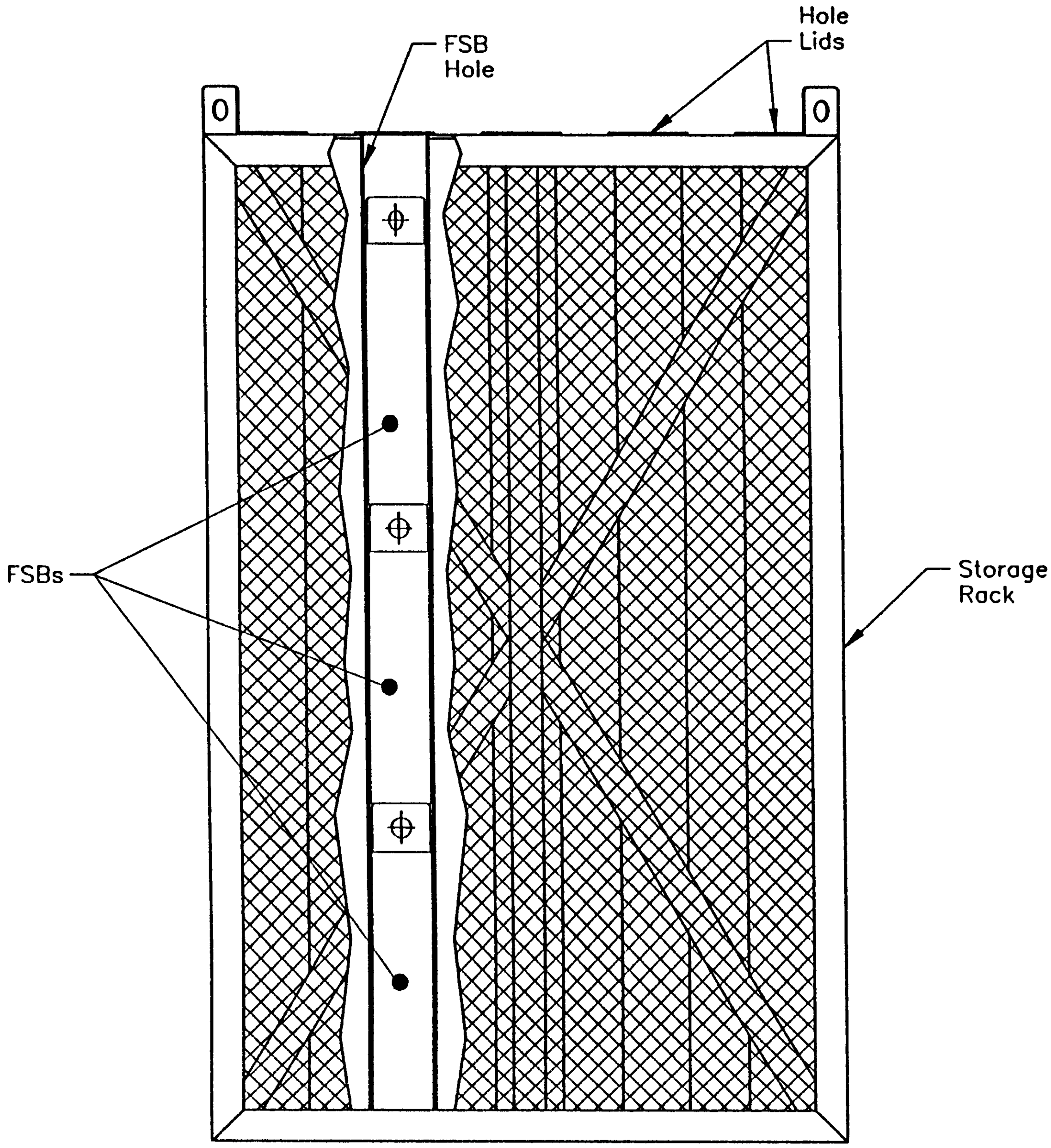


Figure F-38 Stacked Fuel Storage Buckets in Storage Rack

The staff required to operate the wet storage facility is estimated to be a maximum of 30 when 24-hour-a-day fuel loading is being performed, with only occasional maintenance visits by administrative personnel for operation. Potential radiological consequences are extrapolated from other operating wet storage facilities and are discussed in Section F.4.

No high-level radioactive waste is expected to be generated by the wet storage facility. Low-level solid radioactive waste generated over the 40-year life of the facility is expected to be about 488 m^3 (17,200 ft^3). Nonradioactive solid waste generated over the facility's life is expected to be about 300 m^3 (10,594 ft^3). No nonradioactive air emissions are expected to be generated by this facility. Table F-20 summarizes the parameters for the facility.

The cost to construct a wet storage facility with a staging area sufficient to unload, characterize, can, and transfer the spent nuclear fuel to the storage area is estimated to be $449 million. This cost may include some duplicate facilities and equipment present in both the staging facility and the rest of the wet storage facility which were costed separately. The annual operating cost for this facility is estimated to be $23.3 million during the period of handling the spent nuclear fuel and $3.5 million during the period of storage. The cost estimate for the facility is based on a cost report prepared by Idaho Inc. (EG&G, 1993).

F.3.3 Site-Specific Facilities Proposed for Foreign Research Reactor Spent Nuclear Fuel Storage and Management

F.3.3.1 Savannah River Site

RBOF

The Savannah River Site has proposed the use of its RBOF, which is also designated as Building 244-H (DuPont, 1983a and 1983b; Shedrow, 1994a and 1994b; WSRC, 1994a; Claxton et al., 1993; DOE 1993c). The RBOF is a 30-year-old steel and concrete block structure that contains several water pools that have been used for the storage of spent nuclear fuel including foreign research reactor spent nuclear fuel since approximately 1964.

The RBOF facility is located in H-Area on 0.8 ha (2 acres) of land about 397 m (1,300 ft) west of the 221-H Canyon building. A railroad track terminates within the facility, and a roadway surrounds it for access by trucks. The RBOF Building (244-H) is about 42 m (139 ft) wide and 45 m (148 ft) long and contains water-filled basins. The basin area extends below grade to a maximum depth of 13.7 m (45 ft), the roof over the 91 metric ton (100 ton) crane bay is about 13.7 m (45 ft) above grade, and most of the remainder of the roof is at an elevation of 4.6 m (15 ft).

The building consists of seven main sections separated by partition and shielding walls. A ventilation system is provided to exhaust any airborne particulate contamination through filters. The basins, cubicles, and shielding walls are made of reinforced concrete. Most of the above-grade structure consists of standard structural steel shapes with an exterior wall of Transite™ (registered trademark of Johns-Manville Co.). The walls are insulated with Fiberglas™ (registered trademark of Owens-Corning Corp.).

The basin, or working area, of the building has an inner wall of Transite™ to prevent water damage to the insulation from condensation. The disassembly and inspection basins are separated by an inner concrete block wall, and the repackaging basins are enclosed by concrete block walls.

Table F-20 Summary of Wet Storage Parameters for Foreign Research Reactor Spent Nuclear Fuel

Construction Phase:	
Disturbed Land Area	2.8 ha (7 acres)
Facility: Size (Area) Concrete Steel	 3,800 m^2 (41,000 ft^2) 12,400 m^3 (16,260 yd^3) 3,100 metric tons (3,443 tons)
Soil Moved	18,000 m^3 (24,000 yd^3)
Equipment Fuel	600,000 l (159,000 gal)
Construction Debris/Waste	2,600 m^3 (10,300 yd^3)
Work Force	157/yr average, 184 peak
Duration (Years)	4 years for construction, 1.5 years for design
Capital Cost	$449 million[a,b]
Operation Phase:	
Electricity	1,000 - 1,500 MW-hr/yr
Water	2.7 million l/yr (720,000 gal/yr) during receipt, 1.5 million l/yr (409,000 gal/yr) thereafter
Wastestreams	
High Level Waste	none
Transuranic Waste (TRU)	none
Solid Low Level Waste	16 m^3/yr (580 ft^3/yr)
Waste Water	1.59 million l/yr (420,000 gal/yr) during receipt, 0.4 million l/yr (109,000 gal/yr) thereafter
Staff (Full-Time Equivalents)	30
Annual Operating Cost	$23.3 million during handling $3.5 million during storage[a]

Source: (Dahlke et al., 1994)

[a] *Cost estimates are in $1993 (EG&G, 1993)*

[b] *The cost may include duplicate equipment costed in both the staging facility and the wet storage facility*

The two storage pools are between 6.7 and 8.8 m (22 and 29 ft) deep, and approximately 70 percent of their capacity is filled with a variety of fuel types, including aluminum-clad fuel with a ^{235}U enrichment up to 93.91 percent. Subcriticality is maintained by appropriate fuel spacing [center-to-center fuel spacing in these racks currently varies between 23 and 65 cm (9 and 25.5 in) depending on the specific rack] in the storage racks, since no neutron absorption material is used in the pool water. Rack height is 3.4 m (11.17 ft), but some fuel protrudes above the top of the racks. Some of this fuel has been stored at the RBOF for as long as 15 years without any significant degradation. These aluminum storage racks have been present in the pools for 30 years without degradation. Figure F-39 shows the floor plan, and Figure F-40 displays an elevation view.

The RBOF includes specific design, operating, and maintenance procedures for the receipt of a wide variety of fuel types and casks, including damaged fuel elements. The RBOF has the facilities and experience in all aspects of spent nuclear fuel receipt including cask wash, fuel unloading, fuel transfer, fuel storage, fuel inspection, fuel disassembly, and fuel repackaging.

The RBOF pools have a stainless steel bottom and epoxy-coated walls. Pool walls are made of reinforced concrete that varies in thickness from 0.9 m (3 ft) at the top of the pool to 2 m (6.5 ft) at the lower elevations, and the pool floor is a stainless steel liner over a 91-cm (3-ft) thick reinforced concrete slab. Most of the pools have a 6.4-mm (0.25-in) thick stainless steel liner on the floor. The disassembly, inspection, and repackaging basins also have a 3.2-mm (0.125-in) thick stainless steel liner on the walls. Pools or basins are connected by transfer canals with underwater doors.

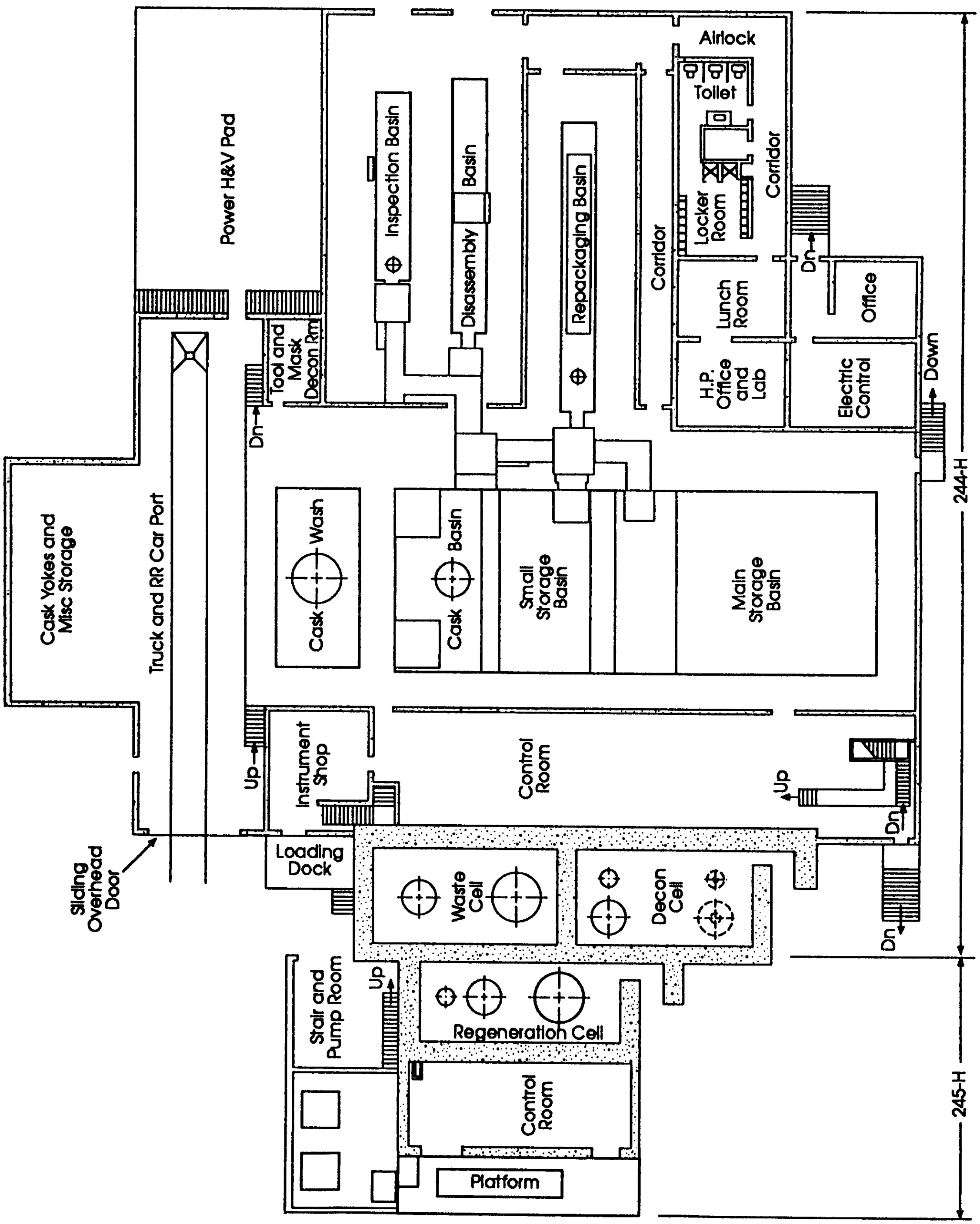


Figure F-39 Plan View of the RBOF

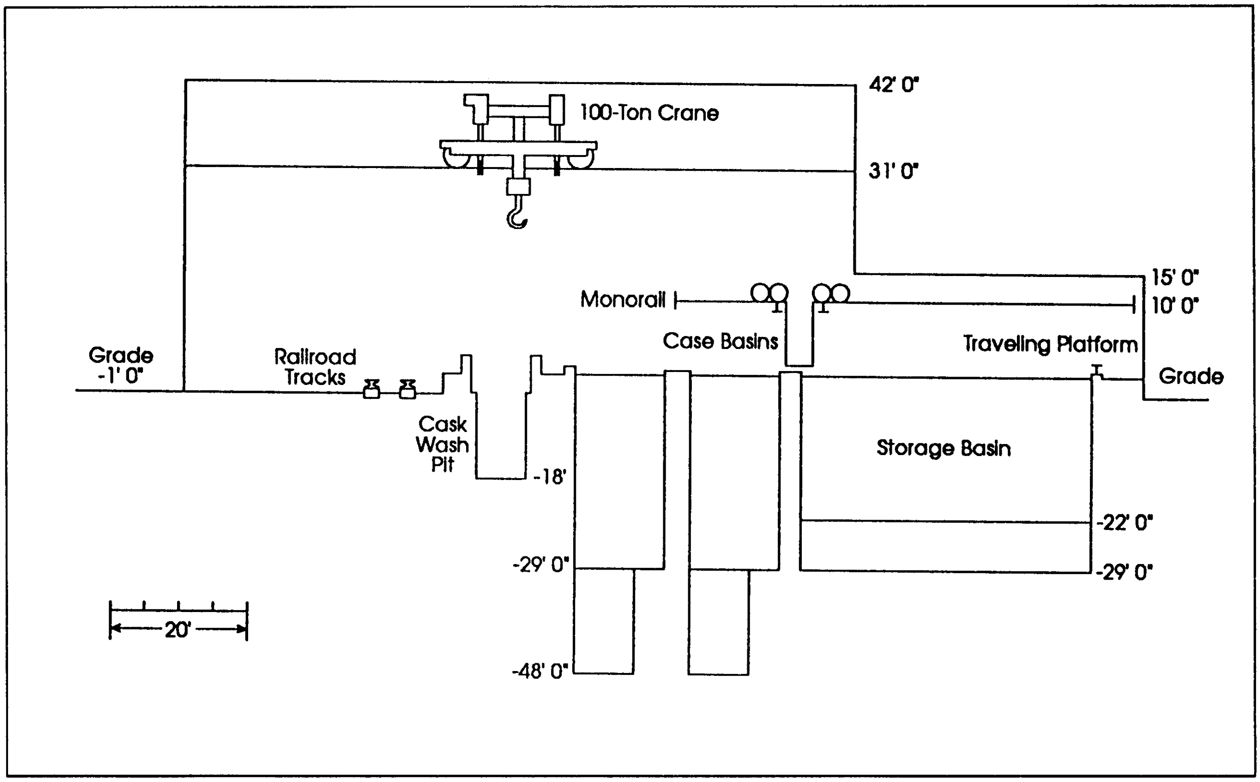


Figure F-40 Elevation Schematic of the RBOF (Facing East)

Water from any basin can be pumped through a filter-deionizer and then returned to the basin as purified water with a conductivity in the range of 0.5 to 1.5 µmhos/cm. In addition, the activity level of the water, which is typically in the range of 0.5 to 1.5 x 10^{-4} µCi/ml, is reduced to less than 0.05 x 10^{-4} µCi/ml by this process. The normal inventory of activity in the approximately 1,700,000 l (450,000 gal) of total basin water is thus 0.1 to 0.3 Ci. For typical flow rates of 454 l/min (120 gal/min), the deionizer processes approximately 1.1 x 10^{8} l (3 x 10^{7} gal) during 6 months of service and may contain about 20 Ci of radioactivity (mostly as cesium) when regeneration is required. The deionizer is typically regenerated every 4 to 5 months. The "Porostone" (aluminum oxide) filter that precedes the deionizer is normally backflushed and recoated with filter-aid whenever a significant pressure drop occurs, which, in practice, is about three to four times per month. When the filter tubes become plugged, they are chemically treated with oxalic acid and sodium hydroxide to open the pores of the filter. This occurs once or twice per year. The purification system maintains excellent chemistry, with mercury and copper kept below two parts per billion, and iron and aluminum maintained below 2 parts per million (ppm). Chloride is maintained below 10 parts per billion.

In the event of an interruption of normal power to the RBOF, critical equipment essential for maintaining personnel safety and containing radioactivity are automatically supplied with emergency power from a 12.5-kilovolt amps, 10-kilowatts, 460-volts gasoline-driven generator.

The building ventilation system serves to minimize airborne radioactivity both inside and outside the facility because of the "once-through" airflow system and the use of High Efficiency Particulate Air filters on the building exhaust. Because of the once-through airflow, activity levels do not tend to increase with time within the building, and the High Efficiency Particulate Air filters serve to effectively remove particulate radioactivity that would otherwise be released to the atmosphere. In addition, the facility is maintained at less than atmospheric pressure so that all building air will be filtered before release. The basin areas are also kept at a lower pressure than the rest of the building, and a separate ventilation system supplies air to the control room, offices, and change room. All air, after filtration, is discharged through a 1.5-m (5-ft) diameter by 16.2-m (53-ft) high stack. The exhaust system for the process vessels in the Waste Cell and the Decontamination Cell is similar to the building system, but is separate and employs acid resistant components. This exhaust is discharged through a 25-cm (10-in) diameter by 16.2-m (53-ft) high pipe.

The 91-metric ton (100-ton) capacity bridge crane travels on a 27.4-m (90-ft) long runway located 9.4-m (31-ft) above grade which permits access to the carport, the cask wash pit, and the cask basin. It is used to handle transportation casks, cask lids, cask basin shims, and a semi-remote impact wrench.

The twin hook crane consists of two 45-metric ton (50-ton) capacity hoist trolleys, which can be arranged for independent travel, or which can be electrically locked to provide for operation as a single unit. Load clearance above the 1.1-m (3-ft 6-in) high cask basin railing is 8.1-m (26-ft 6-in). The bridge crane is pendant-operated from a walkway on the west side of the basins.

A 2.7-metric ton (3-ton) hoist, suspended from a monorail on the south girder of the bridge, is used in the handling of a semi-remotely operated impact wrench. The other 2.7-metric ton (3-ton) hoist is used primarily for the handling of yokes and other ancillary equipment in the yoke storage area adjoining the carport.

Brakes on the cranes and hoist are applied automatically in the event of a power outage.

Two small bridge cranes, one motorized and one manually operated, are employed over the repackaging basin. Both have a load capacity of 2.7 metric tons (3 tons).

The RBOF includes a High Efficiency Particulate Air heating, ventilation, and air conditioning system and maintains subatmospheric pressure within the building to minimize environmental releases of radionuclides. Automatic atmospheric isolation is actuated by activity level monitors inside the RBOF. The RBOF is also equipped with groundwater monitoring for detecting leakage from the pool confinement boundary.

Analysis of the RBOF was performed and included an evaluation of the reliability of process equipment and controls, administrative controls, and engineered safety features. The evaluation identified potential scenarios and radiological consequences. Risks were calculated in terms of 50-year population dose commitment per year (person-rem per year) to the onsite staff and to an individual at the plant boundary. Risk is defined as the product of the expected frequency of a release and the consequences of the release. Consequences are expressed in terms of dose commitment to onsite and offsite populations surrounding the release point.

An evaluation of the RBOF as a potential storage site for foreign research reactor spent nuclear fuel indicates a number of problem areas. The current cask handling capacity of the RBOF is approximately one cask per week. This capacity is based upon facility operations at two shifts per day, 5 days per week. The cask handling capacity could be increased, perhaps to as much as 84 casks per year, if facility operations were expanded to around-the-clock (3 shifts per day), 7 days per week. However, considering that shipments out of the RBOF also require cask handling, the net receipt capacity of the RBOF is practically limited to four casks per month. This capacity would not be sufficient for the potential foreign research reactor spent nuclear fuel cask receipt rate of ~60 casks per year. If the RBOF were used for the receipt and loading of dry storage canisters, its receipt rate could be reduced by half. Only ~1,000 fuel storage spaces are available at the RBOF. Consolidation of the spent nuclear fuel might open an additional 1,425 spaces, but this is much less than that required for the number of foreign research reactor spent nuclear fuel elements under consideration in this EIS. The Savannah River Site has proposed movement of other spent nuclear fuel to the reactor storage basins, and use of dry storage for foreign research reactor spent nuclear fuel.

The DOE Spent Fuel Working Group Report has identified a number of vulnerabilities at the RBOF, including insufficient training, inadequate tornado missile protection, no seismic qualification, lack of water leak detection system, and no up-to-date and approved Safety Analysis Report (DOE, 1993b). It should be noted that a system description and a Safety Analysis Report for the RBOF do exist and were published in 1983. Current recommendations are to address and correct these problems by FY 1996 (DOE, 1993b; Taylor et al., 1994). The 30-year age of these pools may also require analyses to determine the remaining safe lifetime without significant replacement or design modifications.

Reactor Disassembly Basins

Savannah River Site has also proposed the use of one or more of its reactor disassembly basins for Phase 1 storage of foreign research reactor spent nuclear fuel (Shedrow, 1994a and 1994b; Taylor et al., 1994). All of these basins were constructed in the early 1950's and became operational in the mid-1950's. The disassembly basins are similar to each other and are briefly described in the sections that follow, using the L-Reactor disassembly basin as an example.

The L-Reactor performed the basic function of irradiating elements in a heavy water moderated and cooled reactor for the purpose of supplying special nuclear materials for national defense, medical, and research applications. The Savannah River Site production reactors are not currently operating. The disassembly area of the Savannah River Site Production Reactors was designed to serve as a processing area for reactor target and fuel assemblies. This processing included removal of decay heat, disassembly of components, short term storage of fissile product material, and cask loading operations. Total residence time from reactor discharge to shipment to the separation areas was typically 12 to 18 months.

The disassembly basin is arranged into three major sections: the machine basin, the vertical tube storage basin, and the transfer area (Figure F-41). The machine basin and vertical tube storage basins are divided ... following interconnected basins:

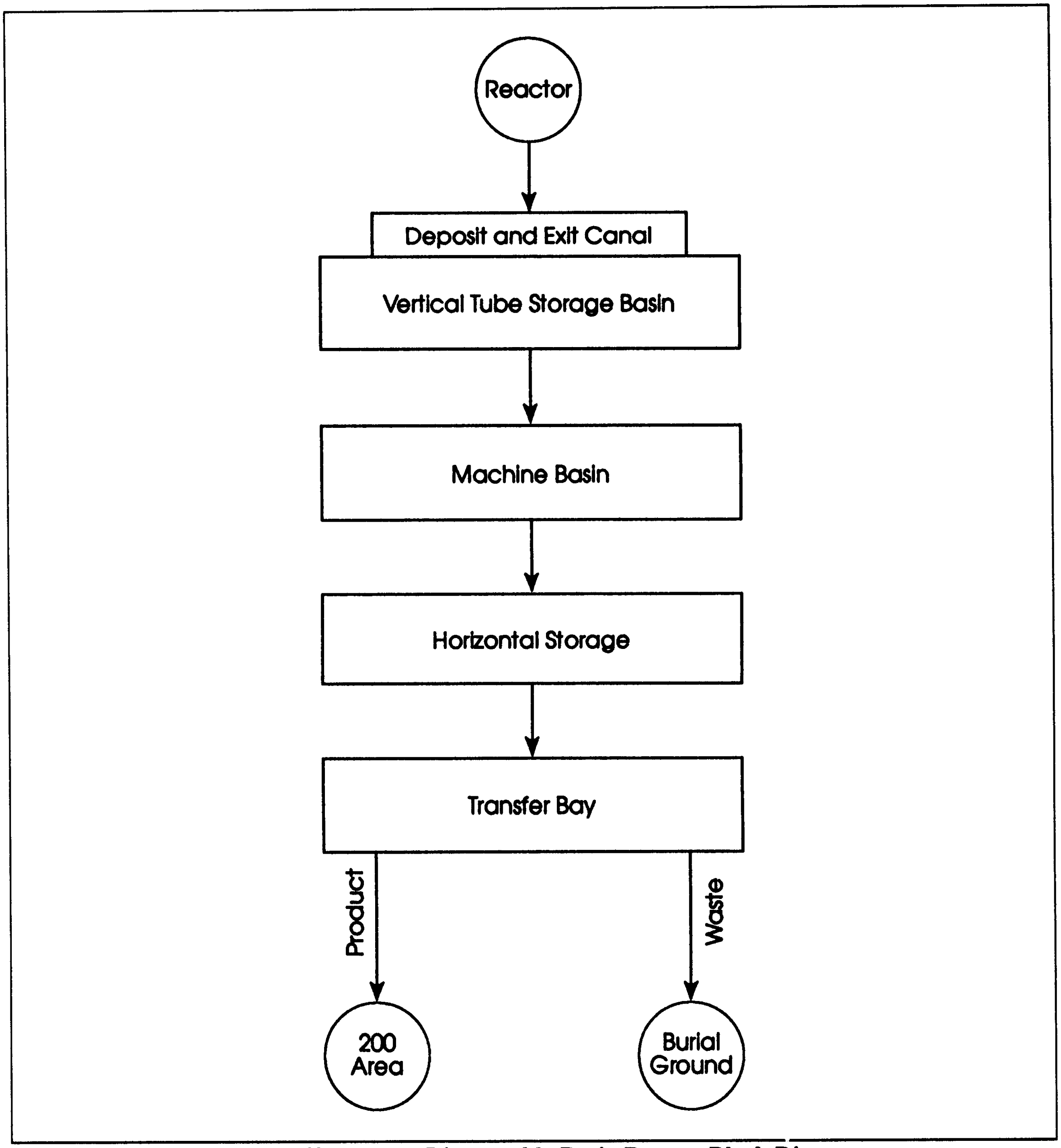


Figure F-42 Reactor Disassembly Basin Process Block Diagram

and transferred to the disassembly area side of the Deposit and Exit canal. The assemblies were transferred to hangers suspended from overhead monorails and were initially stored in the vertical tube storage area for 3 to 8 months. Fuel and target assemblies were moved to the machine basin area where they were disassembled. Target material was placed in stainless steel buckets and then stored in the bucket

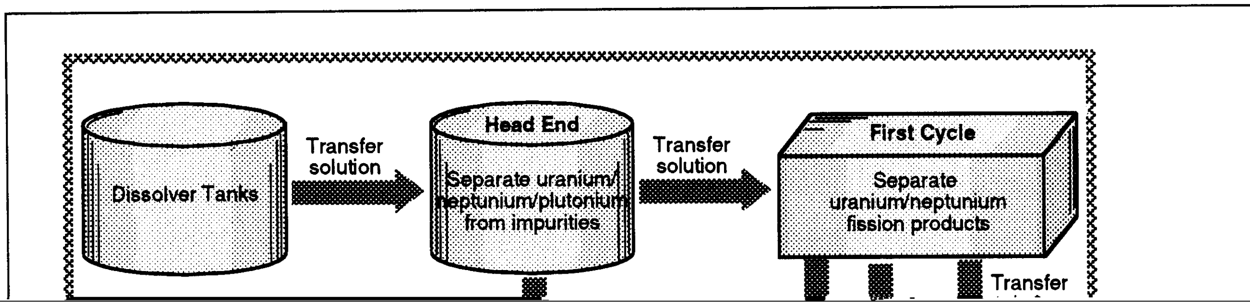
Use of a disassembly basin for foreign research reactor spent nuclear fuel would require continuous demineralizer treatment for water quality, and new storage racks. Existing heat exchanger systems can remove upwards of 6,800 kilowatts (24 x 10^6 BTU/hr), which should far exceed the 240-1,000 kilowatts heat generation rate of foreign research reactor spent nuclear fuel (i.e., 10 to 40 Watts per element). These changes would allow each basin to accommodate approximately 20,000 elements.

[illegible]

two water-filled basins which are designated the scrap pit and the transfer pit. Irradiated material ready for shipping is transferred from horizontal storage to the transfer bay. Transportation casks are moved to and from the transfer pit and irradiated material placed in the cask using hoists mounted on the monorail system. Transportation casks can be transported to and from the reactor areas by tractor trailer or railroad. Trailers or railcars are positioned inside the transfer pit and casks are lifted and transported into/out of the basin using an 85/30 ton overhead crane.

Over the course of the site's history, at least 10 different casks have been used for various applications, many of which are still available for use pending proper inspection and maintenance. Two types are now used for most, if not all, disassembly work. EP-85 is a 63.5-metric ton (70-ton) fuel and target transport cask and EP-383 is a 13.6 metric ton (15-ton) cask used to move scrap to the burial ground.

The transfer area cranes would have to be modified to accommodate the different casks used for offsite shipments. These changes would allow a disassembly basin to receive up to seven casks per month in addition to the projected Savannah River Site shipping requirements.


Dissolver Tanks
Transfer solution
Head End
Separate uranium/neptunium/plutonium from impurities
Transfer solution
First Cycle
Separate uranium/neptunium fission products
Transfer

currently stored in tanks within the facility. Future missions for the facility are still being analyzed. For foreign research reactor spent nuclear fuel, H-Canyon could be used for chemical separation and blending down of HEU to LEU material.

F.3.3.2 Idaho National Engineering Laboratory

On October 17, 1995, litigation with the State of Idaho was settled by stipulation of the parties and entry of a consent order. This settlement would provide for the transportation of up to 61 shipments of foreign research reactor spent nuclear fuel to the Idaho National Engineering Laboratory prior to the year 2000, if DOE and the Department of State choose to adopt a policy of accepting such spent nuclear fuel. After the year 2000, additional shipments of such spent nuclear fuel could be made to the Idaho National Engineering Laboratory under the stipulated settlement and consent order. However, the following discussion is to provide a full understanding of the existing capabilities at the Idaho National Engineering Laboratory in light of the fact that this site has been considered as a reasonable alternative to manage the foreign research reactor spent nuclear fuel as in the preparation of the Draft EIS.

Irradiated Fuel Storage Facility

The ICPP-603 includes an underwater fuel storage basin area and the IFSF, which is a remotely operated, dry-vault facility specifically constructed for the storage of graphite fuel from the Fort St. Vrain and Peach Bottom reactors. It was built in 1974 as an addition to the underwater Fuel Storage Facility and contains 636 storage positions. This facility can handle casks weighing up to 55 metric tons (60 tons). Spent nuclear fuel currently stored here is from two commercial high-temperature, gas-cooled reactors (Fort St. Vrain and Peach Bottom), some for the ROVER Nuclear Rocket Program, and some Tory 2C and BER II TRIGA fuel. The IFSF is a good candidate for spent nuclear fuel requiring frequent monitoring because of the ease of visual fuel inspections.

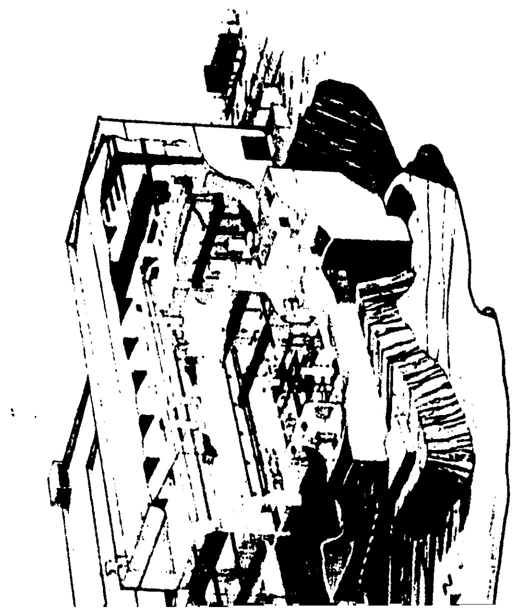
Since the facility can accommodate fuels up to 3.0 m (130 in) in length, all types of foreign research reactor spent nuclear fuel under consideration in this EIS could be handled. Transfer cart modifications would be needed for the proposed foreign research reactor spent nuclear fuel transportation casks, since the existing transfer cart only has the capability to handle the Rover cask, the Fort St. Vrain cask, and the Peach Bottom cask. New fuel handling tools, such as a new can grapple, would be needed. New cell preparations and work stations would also be needed. Sipping, unloading, canning, sealing, and leak checking equipment would need to be added to the cell. A 14 metric ton (15 ton) crane is present in the vault room for fuel handling. Visual inspection and gamma spectroscopy could readily be performed in the existing vault room.

The vault room is 7 m by 7.1 m by 6.6 m (22 ft 10 in by 23 ft 3 in by 21 ft 6 in) high, and the storage room is 437 m^2 (4,700 ft^2). Loaded fuel cans can be transferred from the vault room to the storage area by a shuttle bin. The vault room is being reanalyzed structurally to validate its capability to meet the current seismic requirements of 10 CFR 72. Recent reliable data regarding the effectiveness of the filtering and ventilation systems must be obtained in order to assess the amount of radionuclides that may be vented into the outside air. The cost to add the required capabilities to the IFSF for storage of foreign research reactor spent nuclear fuel is approximately $5 million.

It should be mentioned that, although this facility was originally constructed to accommodate the Fort St. Vrain High Temperature Gas-Cooled Reactor graphite fuel, it will not be used for this purpose because of the October 16, 1995 Settlement Agreement with the State of Idaho that declared that the Fort St. Vrain fuel will not be brought to the Idaho National Engineering Laboratory for interim storage. Public Service of Colorado has obtained a 10 CFR 72 license to store all of the fuel in the Foster Wheeler modular dry vault facility built adjacent to the reactor site. That modular dry vault is currently completely loaded with Fort St. Vrain fuel, and the reactor is being decommissioned. Availability of the IFSF for foreign research reactor spent nuclear fuel will be dependent on decisions to consolidate spent nuclear fuel from other Idaho National Engineering Laboratory facilities at this facility.

Approximately 300 positions are available in the IFSF dry storage facility for foreign research reactor spent nuclear fuel. Preparations to receive the foreign research reactor spent nuclear fuel could be completed as soon as calendar year 1997. However, many activities are already scheduled for this facility. A new canning station for spent nuclear fuel from the ICPP-603 basins is being constructed in the handling cave, and the canning will then be accomplished. Other spent nuclear fuel management activities being

FERMI movements fuel from ICPP-666. Naval fuel inspection sample receipts from the Expended Core Facility are scheduled. The Peach Bottom fuel in the ICPP-749 facility and Rover fuel ash transfers from a shutdown fuel processing facility are also being considered for repackaging in the IFSF handling cell, and the Idaho National Engineering Laboratory spent fuel consolidation activities are scheduled to begin within 2 years. Detailed facility usage schedules have been drafted to demonstrate how the foreign



ssing Plant-666

could be received because most of the ICPP-603 fuel transfers to Fluorinel Dissolution and Fuel Storage (FAST) will have been completed. By the end of 1999, a total of 3,600 fuel elements could be received under this scenario. This schedule is predicated on the assumption that resolution of Idaho National Engineering Laboratory facility vulnerabilities and Naval and Advanced Test Reactor fuel receipts have a higher priority than foreign fuel receipts.

Idaho Chemical Processing Plant-666 Fluorinel Dissolution Process

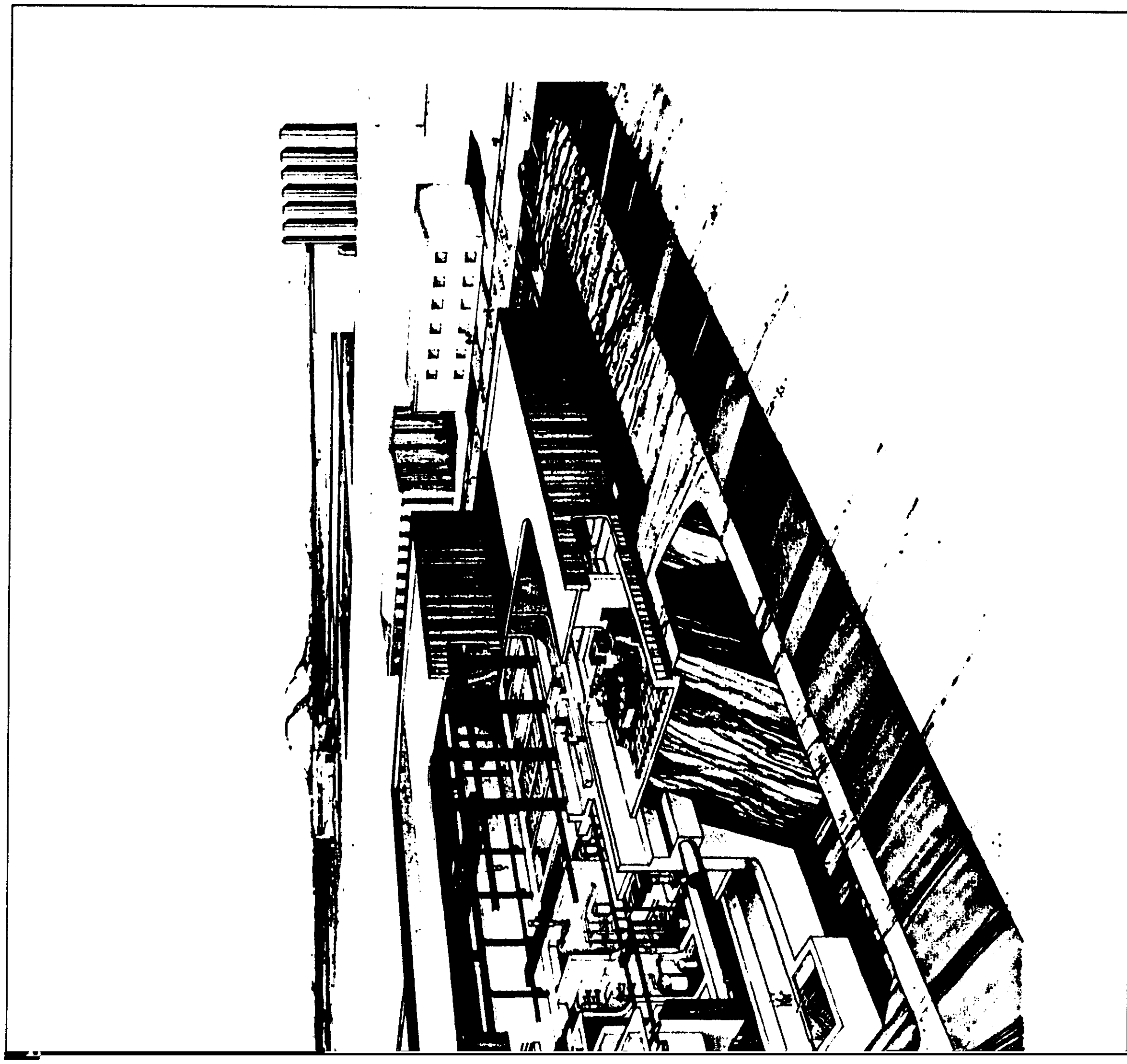
The conversion of the ICPP-666 Flourinel Dissolution Process cell for canning of spent nuclear fuel without removal of any existing equipment or decontamination of the cell is proposed as a low-cost option to prepare foreign research reactor spent nuclear fuel for dry storage. Only minor modifications would be made to the cell, so as to preserve the current dissolution capability for possible future use. This option would utilize the Flourinel Dissolution Process cell, which currently has remote fuel handling, sampling, and waste load-out capabilities, as well as a connection to the ICPP-666 Fuel Storage Area, for fuel inspection, stabilization, and packaging for interim dry storage.

A potential disadvantage is noted. The equipment inside the Flourinel Dissolution Process cell is contaminated, and radiation fields are too high for manned entry. Retaining the dissolution cell equipment will make it impossible to adequately clean the cell to allow personnel entry. For this reason, equipment requiring installation within the Flourinel Dissolution Process cell must be assembled outside the cell and installed remotely with the in-cell crane and master-slave manipulators. Preventative and corrective maintenance of the equipment inside the cell would be done remotely. The modular design of the components would facilitate removal and replacement.

No general Flourinel Dissolution Process utility upgrades, such as electrical power or ventilation, would be required. Piping services could be added to support the vacuum drying and inert gas backfilling functions envisioned to meet dry storage requirements. All of the Reduced Enrichment for Research and Test Reactors (RERTR) program spent nuclear fuel could be accommodated by the existing transfer tunnel and transfer cart. Modifications of the existing fuel shear tools, or new ones, could be acquired to shear larger objects such as cans and lids.

Fuel Processing Restoration

Another structure that may represent an option for fuel storage is the Fuel Processing Restoration building (Figure F-45) that was constructed to house the Fuel Processing Restoration process. It is approximately 56.4 m (185 ft) long and contains shielded, below-grade process cells. These cells vary in dimension, with the nine main process cells measuring 5 to 6 m (16.5 to 20 ft) wide, 10.4 m (34 ft) long, and 12.2 m (40 ft) deep. Fuel racks could be designed to accommodate cans of the type proposed for dry storage in arrays that could contain as many as 17 cans along the 10.4 m (34 ft) axis by 8 cans along the 4.9 m (16 ft) axis, and be 3 cans deep. Airflow through each of these cells could be controlled by dampers in the cell ductwork. Construction of this facility was interrupted prior to completion, and it currently does not include cell ventilation, fire safety equipment, instrumentation, or lighting. These additions would cost approximately $15 million. Several other processes are being considered for use of this facility, and there is no assurance that it would be used for fuel storage. When completed, the building will be seismically qualified, and could physically accommodate approximately 540 canisters per main process cell. Special remote handling tools and techniques would be developed to allow the fuel cans to be inserted into the storage cell. The total estimate to make all necessary conversions is $65 million.



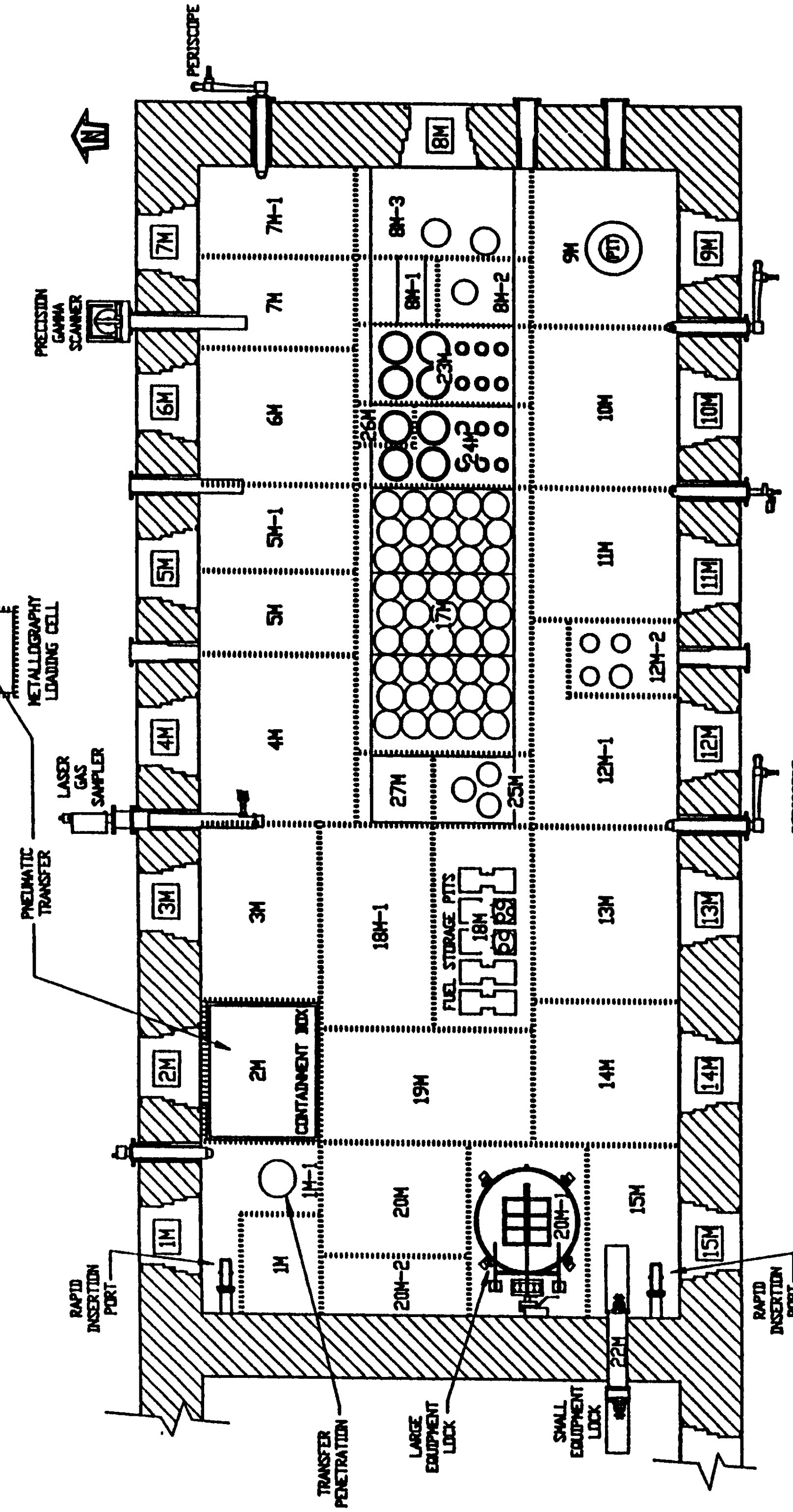
el Processing Restoration Facility (Unfinished)

Hot Fuel Examination Facility

The Hot Fuel Examination Facility is a facility used for examining and storing irradiated fuels from the EBR-II breeder reactor at Idaho National Engineering Laboratory. Figure F-46 presents the layout of the main cell of the Hot Fuel Examination Facility. Although it was not designed for storage, the Hot Fuel Examination Facility could be used to receive, inspect, examine, and transfer foreign research reactor spent nuclear fuel to dry cask storage if it is fitted with an appropriate spent nuclear fuel examination station. The cost of these modifications and the purchase and installation of the dry casks and their equipment would be the principal costs involved.

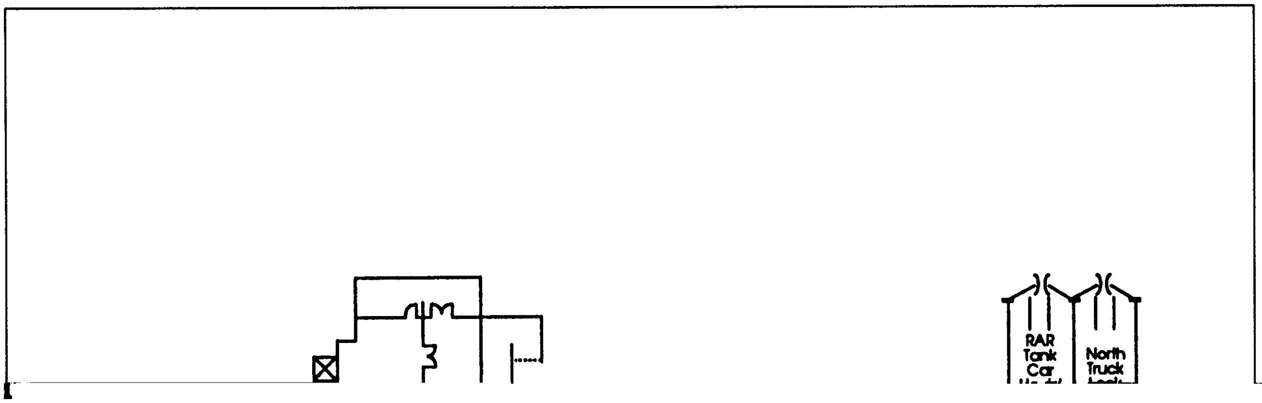
Test Area North-607 Pool, Hot Cell, and Cask Storage Pad

The utilization of the Test Area North-607 facilities is a potential option for receipt and storage of the foreign research reactor spent nuclear fuel. This could be accomplished without significant modification to the hot cell. The hot cell has significant lag capacity for interim storage of foreign research reactor spent nuclear fuel and has most of the equipment necessary for placement of the fuel into dry interim storage casks. There is adequate space for installation of the characterization and conditioning equipment needed for dry storage. There are significant vulnerabilities associated with the underwater storage pool which would need to be corrected if underwater interim storage were desired. The cask storage pad could be easily expanded to accept additional dry storage casks. At the current time, the entire Test Area North area is being planned for shutdown in approximately ten years due to reduced mission needs. If Test Area North had adequate new missions and it was determined to be economical, the Test Area North hot cell pad and storage area would have significant capacity for receipt and temporary storage of foreign research



RESPONSIBLE CHIEF TECHNICIAN	DESIGNATED AREAS
1. METALLOGRAPHIC OPERATIONS GROUP —	1M, 1M-1
2. METALLOGRAPHIC OPERATIONS GROUP —	2M, 25M, 27M
3. IN-CELL EXAMINATION GROUP —	3M, 4M, 5M, 5M-1, 6M, 7M, 7M-1, 17M, 18M-1, 26M
4. SAFETY EXPERIMENT GROUP —	8M-1, 8M-2, 8M-3, 9M, 10M, 23M, 24M
5. S/A DISMANTLING, RECONSTITUTION AND HANDLING GROUP —	11M, 12M-1, 12M-2, 13M, 14M, 18M, 20M, 20M-2
6. FACILITIES AND SYSTEM GROUP —	15M, 19M, 20M-1, 22M

Figure F-46 Hot Fuel Examination Facility Main Cell Layout


RAR
Tank
Car
North
Truck

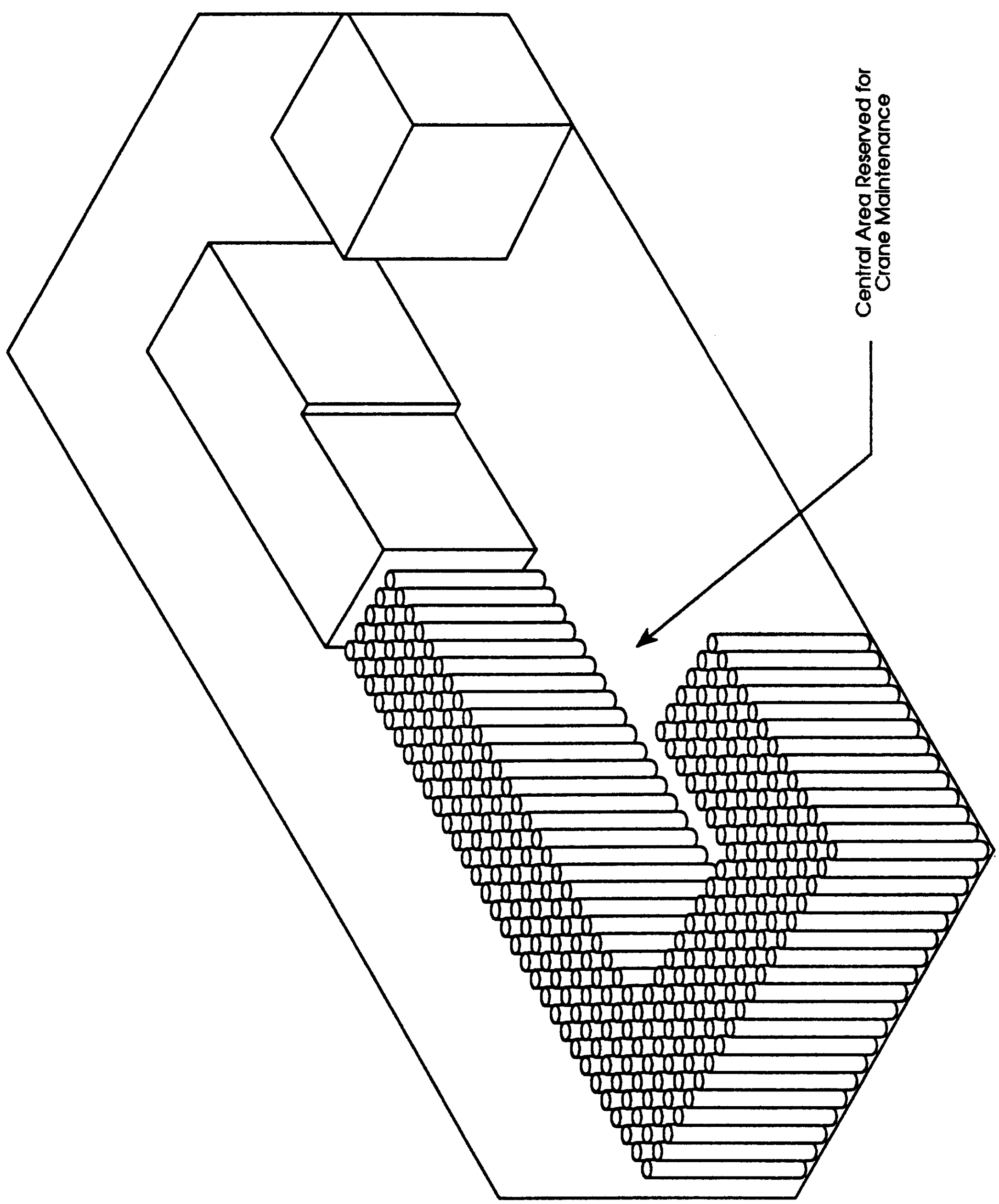


Figure F-48 Potential Use of the Fuel Maintenance and Examination Facility for Fuel Storage
Partial Plan View

The Entry Tunnel is a 9.9 m (32.5 ft) high tunnel below the Shipping and Receiving area floor designed to transfer transportation casks to the Decon and Main Process Cell areas of the FMEF. It includes a 68 metric ton (75 ton) overhead bridge crane. To transport heavier multi-purpose casks when the foreign research reactor spent nuclear fuel would be shipped out of the FMEF in the future, the tunnel would be extended and modified to accommodate an above-grade 114 metric ton (125 ton) crane. The entry tunnel would be extended to connect to the new adjacent storage facility.

The Decon Cell is a room 12.2 m (40 ft) long, 9.1 m (30 ft) wide, and 11.6 m (38 ft) high, with thick concrete shielding. Nine work stations with remote manipulators and viewing windows are part of the design of this cell, although it should be noted that the viewing windows and equipment have not been installed. Access is available through two 2.1 m (84 in) diameter hatches, a 0.8 m (30 in) diameter port, and a 0.3 m (12 in) diameter opening. The Decon Cell includes material handling capability by cranes, manipulators, and hoists ranging from 1.4 to 6.8 metric tons (1.5 tons to 7.5 tons). The Decon Cell would be used to inspect foreign research reactor spent nuclear fuel that has been unloaded from transportation casks and to subsequently load this spent nuclear fuel into storage baskets. Each basket holds three foreign research reactor spent nuclear fuel elements. After basket loading, four baskets would be stacked into a 4.6 m (15 ft) high stainless steel canister. The canisters would be moved to the adjacent storage facility using a Transfer Tunnel, which is equipped with a cart.

The Main Process Cell represents a potential storage location for foreign research reactor spent nuclear fuel at the FMEF (Figure F-47). This room is 30.5 m (100 ft) long, 12.2 m (40 ft) wide, and 11.6 m (38 ft) high with concrete walls either 1.2 or 1.5 m (4 or 5 ft) thick, depending on the concrete's density. The Main Process Cell design includes two 4.5 metric ton (5 ton) bridge cranes and two 1.4 metric ton (1.5 ton) electro-mechanical manipulators.

A zoned heating, ventilation, and air conditioning system with negative differential pressure, redundant cooling systems, and staged multiple High Efficiency Particulate Air filters provides decay heat removal and protection from environmental releases of radioisotopes. This system provides for flow, by negative air pressure differential, from the least contaminated zones to the most contaminated zones, thereby maintaining individual zone relative contamination potential. Supply air is drawn from tornado-hardened and seismically qualified intake shafts and dampers. All heating, ventilation, and air conditioning equipment required to supply high contamination zones is designed as Seismic Category 1. After multiple High Efficiency Particulate Air filtration and monitoring for radioactivity, heating, ventilation, and air conditioning exhaust air is released from a seismically qualified reinforced concrete 35.7 m (117 ft) tall stack.

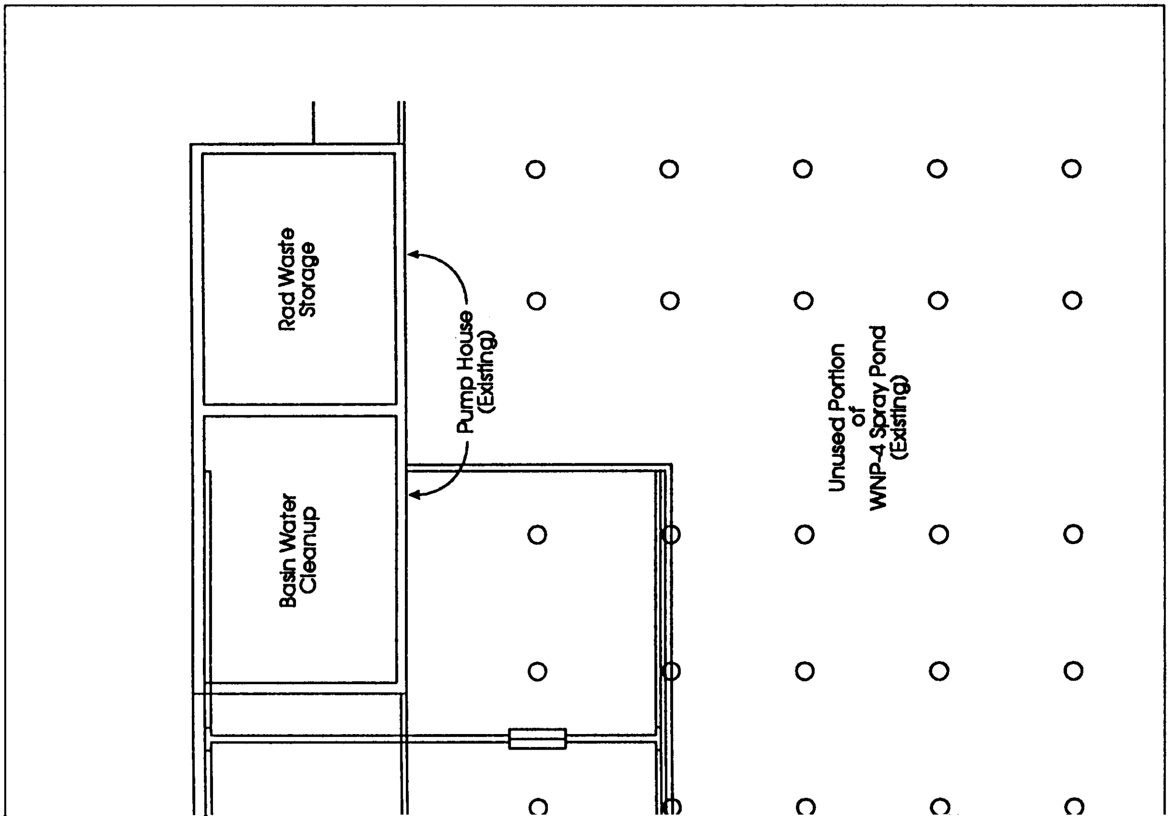
The FMEF is provided normal power by two separate 115 kV electric power supply lines from the Bonneville Power Administration. Emergency power is provided by two 100 percent redundant 900-kilowatt gas turbines which, along with their seismically qualified support and fuel oil systems, are capable of 24 hours of continuous operation. An Uninterruptible Power Supply, consisting of two 150-kVA lead calcium batteries, can provide full load for 30 minutes. Emergency generators require 2 minutes to start up and produce rated power.

A number of modifications would be required for the FMEF to be used as a storage facility for foreign research reactor spent nuclear fuel. They can be categorized as: addition of a 114 metric ton (125 ton) crane, railroad tunnel extension, and storage rack canisters. Even with these modifications, the FMEF does not have sufficient space to store 23,000 foreign research reactor spent nuclear fuel elements, but it could be used as an unloading and support facility for an adjacent dry vault storage facility. Costs for the necessary modifications to the FMEF have been estimated to be approximately $32 million. The adjacent dry storage facility is estimated to cost an additional $100 million. It should be noted that the FMEF is

[redacted]
reactor spent nuclear fuel.

F.3.3.3.2 Washington Nuclear Plant-4 Spray Pond Wet Storage

The Washington Nuclear Plant-4 Spray Pond is a nuclear safety-related structure that was originally designed for decay heat removal following a Loss of Coolant Accident at the Washington Nuclear Plant-4 commercial nuclear power plant. The Washington Nuclear Plant-4 was canceled, but the spray pond structure is essentially complete. This pond is 91.4 m (300 ft) long, 76.2 m (250 ft) wide, and 8.2 m (27 ft) deep, and was designed and built to 10 CFR 50 Appendix B quality assurance standards as a seismic and [redacted]



y Pond (with Modifications) Schematic

F.4 Environmental Impacts at Foreign Research Reactor Spent Nuclear Fuel Management Sites

This section analyzes the environmental impacts associated with the storage of foreign research reactor spent nuclear fuel at the five potential management sites considered in the Programmatic SNF&INEL Final EIS (DOE, 1995g), namely: the Savannah River Site, the Idaho National Engineering Laboratory, the Hanford Site, the Oak Ridge Reservation, and the Nevada Test Site.

The Record of Decision for the Programmatic SNF&INEL Final EIS was issued on May 30, 1995. In

F.4.1 Savannah River Site

If the Savannah River Site is the site to manage all DOE-owned spent nuclear fuel, foreign research reactor spent nuclear fuel would be received and managed at the site until ultimate disposition. If the Savannah River Site is not the site to manage DOE-owned spent nuclear fuel, foreign research reactor spent nuclear fuel could be received and managed at the Savannah River Site until the selected site(s) would be ready to receive the foreign research reactor spent nuclear fuel. The construction of new facilities for managing foreign research reactor spent nuclear fuel is estimated to take about 10 years. Modifications to existing facilities for the same purpose could take less time. This period is referred to as Phase 1. The period following Phase 1 until ultimate disposition is referred to as Phase 2. The amount of spent nuclear fuel that could be received and managed at the Savannah River Site under Management Alternative 1, as discussed in Section 2.2.2, is dictated by the distribution considered in the Programmatic SNF&INEL Final EIS. Accordingly, the Savannah River Site could receive one-half of the foreign research reactor spent nuclear fuel under the Decentralization and the 1992/1993 Planning Basis alternatives, the aluminum-based foreign research reactor spent nuclear fuel under the Regionalization by Fuel Type alternative, the foreign research reactor spent nuclear fuel from eastern ports under the Regionalization by Geography Alternative, or all foreign research reactor spent nuclear fuel under the Centralization Alternative. As discussed in Section 2.6.4.1, the split of foreign research reactor spent nuclear fuel evenly between the Savannah River Site and the Idaho National Engineering Laboratory under the Decentralization and 1992/1993 Planning Basis alternatives in the Programmatic SNF&INEL Final EIS was not considered to have a practical basis, and was therefore not evaluated in detail.

As a potential Phase 1 site under Management Alternative 1, the Savannah River Site would receive and manage foreign research reactor spent nuclear fuel at existing wet storage facilities: RBOF and the L-Reactor disassembly basin. Descriptions of RBOF and the L-Reactor disassembly basin are provided in Section F.3. RBOF is located at the H-Area. It is a facility with provisions for the receipt and storage of irradiated nuclear fuel elements. Since 1963, irradiated spent nuclear fuel elements have been received from offsite reactors and from the Savannah River Site reactors. RBOF provides the capability for underwater unloading of the transportation casks and the handling and storage of the foreign research reactor spent nuclear fuel. The foreign research reactor spent nuclear fuel would be stored in RBOF until its storage capacity is exhausted. Currently, RBOF has space for approximately 1,170 foreign research reactor spent nuclear fuel elements. This capacity could be increased to a total of 2,425 elements by rearrangement and consolidation of existing inventory (O'Rear, 1995).

The L-Reactor disassembly basin is not currently configured for storage of aluminum-based foreign research reactor spent nuclear fuel; however, minor modifications which would provide new storage racks, new handling equipment, safety documentation, etc., along with upgrades in progress to address vulnerabilities associated with water chemistry control, would permit receipt and management of foreign research reactor spent nuclear fuel. Installation of racks equivalent to those in RBOF would provide storage for approximately 20,000 foreign research reactor spent nuclear fuel elements. The modifications to RBOF and L-Reactor disassembly basin are part of the ongoing programs at the site to be performed independent of the proposed action in this EIS.

Between the RBOF and the L-Reactor disassembly basin, there would be sufficient storage capacity and handling capability to accommodate the receipt and management of foreign research reactor spent nuclear fuel during the estimated 10-year period for Phase 1.

An additional option to enhance storage capacity during Phase 1 would be to use RBOF and/or L-Reactor disassembly basin to unload the transportation casks and provide storage capacity in dry storage casks which would be placed near the existing facility. Descriptions of the dry storage casks are provided in Section F.3.

As a Phase 2 site under the basic implementation of Management Alternative 1, the Savannah River Site would continue to receive foreign research reactor spent nuclear fuel beyond Phase 1 in a new dry storage facility that would be constructed at the H-Area. The H-Area is the preferred site among several considered for the construction of new foreign research spent nuclear fuel storage facilities, and is the location assumed for the environmental impacts calculations. An alternative site, equally qualified for construction of new storage facilities is located on a ridge between the P-Reactor and the Pen Branch watershed as indicated in Section 2, Figure 2-14 of this EIS (Shedrow, 1994a). Foreign research reactor spent nuclear fuel stored during Phase 1 would be transferred to the new facility and would be stored there for an additional 30 years until ultimate disposition. The dry storage would encompass a number of designs, examples of which were provided in Section 2.6.5.1.1 and in Section F.3.

The analysis of environmental impacts from the management of foreign research reactor spent nuclear fuel at the Savannah River Site is based on the above considerations. The analysis options selected do not represent all possible combinations, but a reasonable set which provides a typical, and in many cases, bounding estimate of the resulting impacts.

The specific analysis options are as follows:

1A. The Savannah River Site would receive foreign research reactor spent nuclear fuel during Phase 1 and store it at the RBOF and/or the L-Reactor disassembly basin. For the purpose of this analysis, the amount of fuel to be stored is all foreign research reactor spent nuclear fuel that would be received during Phase 1 (approximately 17,500 elements). The spent nuclear fuel would be shipped offsite at the end of Phase 1.

1B. Foreign research reactor spent nuclear fuel stored under analysis option 1A would be transferred to a newly constructed dry storage facility, where it would be stored until ultimate disposition. Foreign research reactor spent nuclear fuel arriving in the United States after Phase 1 concludes would be received and stored at the new dry storage facility. For the purpose of this analysis, the amount of spent nuclear fuel that would be stored would be all the foreign research reactor spent nuclear fuel eligible under the policy (22,700 elements).

The implementation alternatives of Management Alternative 1 for managing foreign research reactor spent nuclear fuel in the United States, as discussed in Section 2.2.2, introduce additional analysis options that would be considered for the Savannah River Site as follows:

- Under Implementation Subalternative 1a (Section 2.2.2.1), the amount of spent nuclear fuel to be received in the United States would be reduced to 5,000 elements. In this case, the Savannah River Site would be likely to receive and manage foreign research reactor spent nuclear fuel in existing facilities during the Phase 1 period. The impacts would be bounded by analysis option 1A (above). Impacts of construction and operation of the dry storage facility considered in analysis option 1B would bound those of the facility required to accommodate this amount of fuel. The spent nuclear fuel would either be shipped offsite after Phase 1, or it would be managed along with the rest of the spent nuclear fuel at the Savannah River Site.

- Under Implementation Subalternative 1b (Section 2.2.2.1), the Savannah River Site would receive only HEU from the foreign research reactors eligible under the policy. The amount of HEU would be approximately 4.6 MTHM, representing 11,200 elements. The impacts from the management of this amount of spent nuclear fuel at the Savannah River Site would be bounded by analysis options 1A and 1B above.

- Under Implementation Subalternative 1c (Section 2.2.2.1), the Savannah River Site would receive target material in addition to the foreign research reactor spent nuclear fuel considered under the basic implementation of Management Alternative 1. The receipt and management of this material, which in uranium content represents approximately [illegible] typical foreign research reactor spent nuclear fuel elements, would increase the fuel that would be managed in these facilities would be all the foreign research reactor spent nuclear fuel eligible under the policy (approximately 22,700 elements).

- Under Implementation Alternative 6 (Section 2.2.2.6), DOE and the Department of State would consider chemical separation of foreign research reactor spent nuclear fuel in the United States. As noted in Section 2.3.6, the Savannah River Site is currently limited to chemical separation of aluminum-based foreign research reactor spent nuclear fuel.

Under Management Alternative 2, as discussed in Section 2.3, DOE and the Department of State would assess the management of foreign research reactor spent nuclear fuel in a foreign location which would include an evaluation of foreign reprocessing with acceptance by the United States of the vitrified high-level waste resulting from reprocessing. The waste would be received and managed at the Defense Waste Process Facility at the Savannah River Site. DOE estimates that the total volume of the vitrified high-level waste would be about 2.4 m^3 (8.5 ft^3) and it would fill about 16 European-size canisters. A European-size canister is about four times smaller than the canister used in the Defense Waste Process Facility at the Savannah River Site.

Under Management Alternative 3 (Hybrid Alternative), as discussed in Section 2.4, the Savannah River Site would receive the aluminum-based fuel which would not be reprocessed overseas. This spent nuclear fuel would be processed at the Savannah River Site chemical separation facilities in the same manner as in Implementation Alternative 6 above. The amount of foreign research reactor aluminum-based spent nuclear fuel to be chemically separated would be approximately 12,200 elements, 12.9 MTHM, 79 m^3 (2,600 ft^3).

F.4.1.1 Existing Facilities (Phase 1)

Analysis option 1A utilizes existing facilities that would be ready to receive and store foreign research reactor spent nuclear fuel by late 1995. The environmental impacts from this analysis option include only those related to operations, specifically: socioeconomics; occupational and public health and safety; materials, utilities, and energy; air quality; and waste management. For this analysis, it was assumed that the amount of foreign research reactor spent nuclear fuel to be received at the management site is the maximum, and the receipt rate is uniform at approximately 1,800 elements per year.

F.4.1.1.1 Socioeconomics

Potential socioeconomic impacts associated with analysis option 1A would be attributable to the staffing requirements for existing facilities. Currently, these facilities are being used to store spent nuclear fuel, so any incremental staffing requirements related to foreign research reactor spent nuclear fuel storage would be small. All personnel required for the operation and support of the existing facilities could be acquired from the current work force at the Savannah River Site. Use of the current work force would not result in any net socioeconomic impact relative to baseline employment data. In fact, using the current work force may partially compensate for the decline in employment expected from changes in site mission from 20,000 persons in 1995 to approximately 15,800 persons in 2004 (DOE, 1995g).

F.4.1.1.2 Occupational and Public Health and Safety

Radiological exposures could affect occupational and public health and safety. Possible sources of radiological exposure from the receipt and storage of foreign research reactor spent nuclear fuel include: (1) airborne emissions from incident-free operations; (2) incident-free handling activities; and (3) airborne emissions from accident conditions. Radiological exposures are presented in individual subsections for emissions-related impacts and handling-related impacts. Accident-related impacts are presented in Section F.4.1.3.

Emissions-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Savannah River Site would be attributed to airborne emissions of radioactive material that could be carried by wind offsite. The general public would be too far from the locations where handling activities or storage would take place to receive any dose from direct exposure. Doses were calculated for the maximally exposed individual (MEI), defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the existing storage facility (RBOF and/or L-Reactor disassembly basin) during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.6 of this appendix. Table F-21 summarizes the annual emission-related doses to the public and the associated risks for the MEI and the population at the Savannah River Site during Phase 1 operations.

Table F-21 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage in Existing Facilities at the Savannah River Site (Phase 1)

	MEI Dose (mrem/yr)	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• RBOF (wet)	0.00011	5.5×10^{-11}	0.0057	0.0000028
• L-Reactor Basin (wet)	0.000073	3.7×10^{-11}	0.0046	0.0000023
Storage at:				
• RBOF (wet)	1.2×10^{-9}	6.0×10^{-16}	6.2×10^{-8}	3.1×10^{-11}
• L-Reactor Basin (wet)[a]	0.00036	1.8×10^{-10}	0.022	0.000011

[a] *L-Reactor basin doses are due to existing conditions. The foreign research reactor spent nuclear fuel contribution would be six orders of magnitude lower.*

Handling-Related Impacts: Management site workers would receive radiation doses during handling operations, such as receiving and unloading the transportation casks, transferring the spent nuclear fuel from one facility to another, or preparing the spent nuclear fuel for shipment offsite. Analysis option 1A involves the receipt of 644 shipments of spent nuclear fuel into the existing wet storage facility (RBOF and/or L-Reactor disassembly basin) during Phase 1, and the preparation of 161 transportation casks for offsite shipment at the end of Phase 1. It was assumed that at the end of a 10-year period (i.e., Phase 1), the spent nuclear fuel would have decayed sufficiently to be accommodated in larger capacity transportation casks, such as those currently used in the United States for commercial spent nuclear fuel. For the purpose of this analysis, the transportation casks used for intrasite shipping are assumed to have a capacity four times as large as the capacity of the transportation casks used for the marine transport of the foreign research reactor spent nuclear fuel to the United States. The assumptions and methodology used to calculate the doses to a working crew associated with the handling activities of the spent nuclear fuel are described in Section F.5 of this appendix.

The collective doses that would be received by the members of the working crew and the associated risk were calculated for Phase 1 operations. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculation. However, the upper bound for such a dose would be equal to administrative or regulatory limits at the management site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If one worker received the full 5,000 mrem per year [illegible]

F.4.1.1.5 Air Quality

Nonradiological Emissions: Impact assessments for nonradiological air emissions associated with implementation of the respective spent nuclear fuel management alternatives (excluding construction-related activities) are based primarily on analyses performed for the Programmatic SNF&INEL Final EIS (DOE, 1995g). These analyses were based on the following assumptions and qualifications:

- Air emissions data for wet storage are based upon releases from the RBOF
- All air pollutant sources, except standby diesel generators, are assumed to operate continuously. The standby diesel generators are assumed to operate daily for 1 hour; whereas their actual operation consists of a single monthly test.

Table F-24 lists the annual potential maximum emissions of criteria and toxic air pollutants (in tons per year) attributable to existing facilities. These data indicate little or no difference in pollutant loading between the baseline and the regionalization alternative considered in the Programmatic SNF&INEL Final EIS. The greatest pollutant contribution for criteria air pollutants would be for nitrogen oxides (7.7 metric tons per year, or 8.5 tons per year) and carbon monoxide (1.8 metric tons per year, or 2.0 tons per year). Assuming that the foreign research reactor spent nuclear fuel comprises approximately nine percent of the total spent nuclear fuel managed under the Centralization Alternative of the Programmatic SNF&INEL Final EIS, the incremental and cumulative nonradiological air quality impacts attributable to the storage of foreign research reactor spent nuclear fuel in existing facilities (RBOF, L-Reactor disassembly basin) would be small.

Table F-24 Annual Maximum Emissions of Criteria Air Pollutants Attributable to Foreign Research Reactor Spent Nuclear Fuel Storage at Existing Facilities at the Savannah River Site (Phase 1)

Pollutant	Baseline Wet (tons/yr)[a,b,c]	RBOF (tons/yr)[a,b]	Percent Increase

F.4.1.1.6 Water Resources

The use of RBOF and the L-Reactor disassembly basin for the interim storage of foreign research reactor spent nuclear fuel would not change the current levels of water usage at these facilities. Nor would it change thermal discharges from cooling water or the quantity or quality of radioactive and nonradioactive wastewater effluents.

Viable accidents during this interim storage period could be a release of pool water onto the ground surface or a breach of the liner of the wet storage basins in which the spent nuclear fuel would be stored. These type of accidents have been analyzed for both the RBOF and the L-Reactor disassembly basin in the safety analysis documentation (Dupont, 1983a and 1983b; WSRC, 1995b and 1995c) and the Programmatic SNF&INEL Final EIS (DOE, 1995g). As discussed in the Programmatic SNF&INEL Final evenly distributed over this 4-year period, the annual expenditures would be about $92.5 million. This represents about 7.7 percent of the estimated FY 1995 total expenditures for the Savannah River Site. The relative socioeconomic impact from annual construction expenditures on the region of influence would be small but positive. The annual operations costs from a new dry storage facility are estimated to be $15.6 million for receipt and handling and $.6 million for storage. These costs represent about 1.3 percent and 0.05 percent of FY 1995 total expenditures for the Savannah River Site. The relative socioeconomic impact from annual operation expenditures on the region of influence would be small.

Direct employment associated with construction of new dry storage facility is estimated to be 190 persons. The relative socioeconomic impact from construction employment on the region of influence would be small. In addition, when compared to the projected FY 1995 work force at the Savannah River Site of approximately 20,000 persons, the relative socioeconomic impact of this temporary increase in construction employment would be insignificant. Direct employment associated with receipt and storage operations is estimated to be 30 persons. Upon completion of these activities, direct employment is expected to decrease to eight persons. The relative socioeconomic impact of this increase in operations employment would be insignificant to both the region of influence and the Savannah River Site.

F.4.1.2.1.3 Cultural Resources

There are no known cultural or historic resources located within the two proposed construction locations for a new dry storage facility. Both locations are within an area of low archaeological site density. Activities within this zone would have a low probability of encountering archaeological sites and virtually no chance of impacting large sites with more than three prehistoric components. Neither location has been specifically surveyed for archaeological resources, but this would occur prior to initiation of any construction-related activities.

Three Native American groups have expressed concerns relating to the possible existence on the Savannah River Site of several plant species traditionally used in Tribal ceremonies. These plant species are known to occur on the Savannah River Site, typically in wet, sandy areas such as evergreen shrub bogs and savannas. However, these plants are not likely to be found in the two proposed construction locations because of a lack of suitable habitat.

F.4.1.2.1.4 Aesthetics and Scenic Resources

Construction and operation of a new dry storage facility would not adversely impact aesthetic or scenic resources. A new dry storage facility would not be visible from any onsite or offsite public access roads. Potential soil erosion and dust generation associated with construction-related activities would be controlled by the implementation of best-management practices. Any visibility impacts from fugitive dust generation by construction-related activities should be insignificant and short term. Facility operations associated with the new dry storage of foreign research reactor spent nuclear fuel should not generate any atmospheric emissions which would reduce area visibility.

F.4.1.2.1.5 Geology

There are no unique geologic features or minerals of economic value on the Savannah River Site that would be adversely impacted by site development. Construction of a new dry storage facility would result in localized impacts to surficial soils and would necessitate the clearing and grading of 3.7 ha (9 acres).

Site preparation, land shaping and grading activities associated with construction would present a slight to moderate erosion hazard, which would be controlled and minimized by implementing best-management practices. The operation of the new dry storage facility would have no effect on the geologic characteristics at the Savannah River Site.

F.4.1.2.1.6 Air Quality

Nonradiological Emissions: Potential air quality impacts associated with construction-related activities include the generation of fugitive dust (particulate matter), smoke from earth moving and clearing operations, and emissions from the construction equipment. Sources of fugitive dust include:

- transfer of soil to and from haul trucks and storage piles;
- turbulence created by construction vehicles moving over cleared, unpaved surfaces; and
- wind-induced erosion of exposed surfaces.

Cleared vegetation would be burned at the construction site rather than hauled to a landfill. The open burning of this material is not expected to adversely impact ambient air quality at the Savannah River Site. As shown in Table F-25, air quality impacts associated with construction-related activities would be minimal and compliance with Federal and State ambient air quality standards would not be adversely affected. Therefore, construction activities would not be expected to have any detrimental effect on the health and safety of the general population.

Table F-25 Estimated Maximum Concentrations of Criteria Pollutants at the Savannah River Site Attributable to New Dry Storage Construction

Pollutant	Averaging Time	Ambient Standard[a]	Baseline Concentration[b]	Construction Activities
Savannah River Site Boundary ($\mu g/m^3$):				
• Total Suspended Particulate (TSP)	Annual	75	11	0.002 - 0.003
• Particulate Matter (PM_{10}), Daily	24-hr	150	56	0.1
• Particulate Matter (PM_{10}), Daily	Annual	50	2.7	0.003

[a] *Source: (DOE, 1995g)*

[b] *Baseline values due to actual emissions from all the Savannah River Site sources during 1990 plus sources permitted through 1992*

No nonradiological air emissions would be expected during operation of a new dry storage facility. Any emissions associated with new dry storage would be directly attributable to front-end wet storage activities only.

Radiological Emissions: No radiological emissions would be produced during construction of a dry new storage facility.

Based on fuel drying and storage operations conducted at the Idaho National Engineering Laboratory, potential atmospheric releases from the spent nuclear fuel storage facility would consist of minor amounts of particulate radioactive material and larger amounts of gaseous fission products that could escape from the fuel through cladding defects. The majority of radioactive material responsible for fuel and cask internal surface contamination consists of activation products that plate out on the spent nuclear fuel assemblies during reactor operation. This material is dependent on corrosion of structural materials and generally consists of radionuclides, such as ^{58}Co, ^{60}Co, ^{59}Fe, etc. This contamination activity would have to be controlled during the cask opening and fuel handling operations to prevent internal personnel exposures. Proper facility ventilation (designed to provide airflow from areas of low contamination to progressively high contamination) would help provide contamination control. High-Efficiency Particulate Air filters in the facility exhaust would reduce the airborne effluent quantities of this particulate material to quantities that are well within the prescribed limits.

Cask opening and fuel drying operations may also be responsible for the release of significant amounts of ^{3}H, ^{85}Kr, and minor amounts of ^{129}I. The amounts of these radionuclides released during the cask opening operation depend on the following parameters: (1) the number of spent nuclear fuel clad defects; (2) the spent nuclear fuel material and the diffusion rate of these radionuclides through the fuel matrix for the fuel temperature while in the cask; and (3) the time that the spent nuclear fuel is contained within the cask before opening.

Similarly, for fuel drying operations, the temperature of the drying gas (as well as the parameters discussed above) would cause quantities of ^{3}H, ^{85}Kr, and ^{129}I to be released from the fuel. Charcoal or silver zeolite filters could be used to remove the ^{129}I from the exhaust, but the ^{3}H and ^{85}Kr, being gases, or in a vapor state for the case of tritiated water, would be exhausted to the atmosphere. During spent nuclear fuel storage small amounts of the gaseous/volatile radionuclides are expected to be released to the environment based on the fuel matrix, clad defects, and storage temperature. Release rates would decrease with storage time due to radioactive decay. It is anticipated that the fuel drying operation would be responsible for the most significant release of these gaseous/volatile radionuclides to the environment.

Radiological emissions from the operation of a new dry storage facility were calculated based on the methodology and assumptions discussed in Section F.6. The radiological consequences of air emissions are discussed in Section F.4.1.2.1.11. The annual emission releases from the dry storage facility during receipt and unloading and storage are provided in Section F.6.6.1.

F.4.1.2.1.7 Water Resources

The water usage during construction of a new dry storage facility is estimated to be about 7.75 million l (2 million gal). During operations, annual water consumption would be 2.1 million l (550,000 gal) for receipt and handling and 0.4 million l (109,000 gal) for storage. With an annual average water usage of approximately 88,200 million l (23,300 million gal) for the Savannah River Site, these amounts represent no more than a 0.002 percent increase in annual water usage. Therefore, a new dry storage facility would have minimal impact on water resources at the Savannah River Site.

Best-management practices during construction would prevent sediment runoff or spills of fuels or chemicals. Therefore, construction activities should have no impact on water quality at the Savannah River Site. The impact on water quality during operations would also be negligible. Existing water treatment facilities at the Savannah River Site could accommodate any new domestic and process wastewater streams from a new dry storage facility. The expected total flow volumes at the Savannah River Site would still be well within the design capacities of treatment systems at the Savannah River Site. A new dry storage facility would meet National Pollutant Discharge Elimination System limits and reporting requirements, so no impact on the water quality of receiving streams is expected.

F.4.1.2.1.8 Ecology

Terrestrial Resources: The two proposed locations for new spent nuclear fuel management facilities encompass approximately 60-plus ha (150-plus acres) of undeveloped forest land. Surface vegetation consists primarily of upland pine stands. Loblolly and slash pine dominate, but small pockets of hardwoods (oaks, hickory, sweetgum, and yellow poplar) are also evident. The locations possess suitable habitat for white-tailed deer and feral hogs, as well as other faunal species common to the mixed pine/hardwood forests of South Carolina. The locations contain no suitable habitat for the various threatened and endangered species found on the Savannah River Site. The construction of a new dry storage facility would necessitate the clearing of 3.7 ha (9 acres) and is therefore not expected to significantly affect the terrestrial ecology of the area.

Wetlands: Dry storage of foreign research reactor spent nuclear fuel would not adversely impact wetlands. Although two small wetland areas are located along the southeastern perimeters of the potential storage locations, there is sufficient land area available within these locations to avoid these critical habitats. The implementation of best-management practices to control surface runoff and sedimentation would ensure the protection of wetlands and the aquatic ecosystem during construction activities.

Threatened and Endangered Species: The potential locations contain no suitable habitat for threatened, endangered, or candidate species known to occur on or near the Savannah River Site (DOE, 1995g). The southern bald eagle and wood stork feed and nest near wetlands, streams, and reservoirs, and thus would not be attracted to the highly industrialized foreign research reactor spent nuclear fuel management sites. Red-cockaded woodpeckers prefer open pine forests with mature trees greater than 70 years old for nesting and 30 years old for foraging. It is not believed that this species utilizes the relatively young pine stands (5 to 40 years of age) present within the potential storage locations. The nearest red-cockaded woodpecker colony is located across Upper Three Runs Creek, approximately 3.2 km (2 mi) north of H-Area. DOE has begun consultations with the U.S. Fish and Wildlife Service to determine the potential for endangered species to be affected, as required by the Endangered Species Act. Impacts to threatened and endangered species are not anticipated.

F.4.1.2.1.9 Noise

Noise generated onsite by construction or operation of a new dry storage facility should not adversely affect the public or the Savannah River Site environment. Noise generated by construction would be site specific and short lived. A small number of new construction jobs would be generated, but the resultant temporary increase in worker and materials traffic is expected to be insignificant compared to existing baseline traffic loads. Noise generated by operation would not significantly impact the environment because the facility would be located adjacent to previously developed, industrialized areas. The number of daily freight trains in the region and through the site (approximately 13) are not expected to change as a result of dry storage. There may be a slight increase in truck traffic to and from the potential storage locations, but this is not expected to result in a perceptible increase in traffic noise or any change in community reaction to noise along the major access routes to the Savannah River Site.

F.4.1.2.1.10 Traffic and Transportation

Construction materials, wastes, and excavated materials would be transported both onsite and offsite. These activities would result in increases in operation of personal-use vehicles by commuting construction workers, commercial truck traffic, and in traffic associated with the daily operations of the Savannah River Site. Again, traffic congestion would not be a significant problem. As long as commercial trucks are complying with the Federal and State loading and speed regulations, truck traffic would not significantly damage the roadbed.

Traffic due to operations of a new dry storage facility would not increase site levels because the required workers would be drawn from the existing the Savannah River Site labor force.

F.4.1.2.1.11 Occupational and Public Health and Safety

Emissions Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Savannah River Site would be attributed to emissions of radioactive material that could be carried by the wind offsite. The general public would be too far from the locations where handling activities or storage take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the storage facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.6 of this appendix. Table F-26 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Savannah River Site. Integrated doses for the duration of a specific implementation period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-26 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at Savannah River Site (New Dry Storage)

Facility	MEI Dose (mrem/yr)	MEI Risk (LCF/yr)	Population Dose (person-rem/yr)	Population Risk (LCF/yr)
Receipt/Unloading at: • New Dry Storage Facility	0.00018	9.0×10^{-11}	0.0086	0.0000043
Storage at: • New Dry Storage Facility	0	0	0	0

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask), transferring the spent nuclear fuel from one facility to another, or preparing the spent nuclear fuel for shipment offsite. Analysis option 1B involves the receipt of 644 shipments of foreign research reactor spent nuclear fuel into an existing wet storage facility (RBOF and/or L-Reactor disassembly basin) during Phase 1, the preparation of 161 transportation casks for shipment to a dry storage facility at the end of Phase 1, and the receipt of 193 shipments of foreign research reactor spent nuclear fuel directly from the ports to the new dry storage facility after Phase 1 operations. It was assumed that at the end of a 10-year period, the foreign research reactor spent nuclear fuel would have decayed sufficiently to be accommodated in larger capacity transportation casks, such as those currently used in the United States for commercial spent nuclear fuel. For the purpose of this analysis, the transportation casks used for intrasite shipping are assumed to have a capacity four times as large as the capacity of the transportation casks used for the marine transport of the foreign research reactor spent nuclear fuel to the United States. Doses were calculated for the dry vault and dry cask designs. The assumptions and methodologies used to calculate the doses to a working crew associated with the handling activities of the foreign research reactor spent nuclear fuel are described in Section F.5 of this appendix.

Table F-27 presents the population dose that would be received by the members of the working crew and the associated risk if that working crew handled the total number of transportation casks at the Savannah River Site. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the administrative or regulatory limit at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job to prevent further exposure. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. For this dose, the associated risk of incurring an LCF would be 2.6 percent.

Table F-27 Handling-Related Impacts to Workers at the Savannah River Site (New Dry Storage)

	Worker Population Dose (person-rem)			*Worker Population Risk (LCF)*		
	RBOF/L-Reactor	*New Dry Storage Cask*	*New Dry Storage Vault*	*RBOF/L-Reactor*	*New Dry Storage Cask*	*New Dry Storage Vault*

NA = Not Applicable

F.4.1.2.1.13 Waste Management

Construction of a new dry storage facility at the Savannah River Site would generate approximately 1,800 m^3 (2,400 yd^3) of debris. The annual quantities of waste generated during operations are shown in Table F-29. These quantities represent a very small percent increase above current levels at the Savannah River Site. Existing waste management storage and disposal activities at the Savannah River Site could accommodate the waste generated by a new dry storage facility. Therefore, the impact of this waste on the existing Savannah River Site waste management capacities would be minimal.

Table F-29 Annual Waste Generated from New Dry Storage at the Savannah River Site

Waste Form	*Baseline Site Generation*	*Dry Storage Generation*	*Percent Increase*
High-Level Waste (m^3/yr)	127,400[a]	none	0 percent
Transuranic Waste (m^3/yr)	760	none	0 percent
Solid Low-Level Waste (m^3/yr)	19,750	22[b] 1[c]	0.11 percent[b] 0.005 percent[c]
Wastewater (l/yr)	690,000,000	1,590,000[b] 400,000[c]	0.21 percent[b] 0.06 percent[c]

[a] *Total inventory (m^3) at the Savannah River Site*

[b] *During receipt and handling*

[c] *During storage*

F.4.1.2.2 Wet Storage

Analysis option 1C is associated with the construction and operation of a new wet storage facility or the modification and operation of BNFP at the Savannah River Site (Implementation Alternative 5 to Management Alternative 1). The environmental impacts from the modification of the BNFP would be bounded by the impacts associated with the construction of a new wet storage facility.

F.4.1.2.2.1 Land Use

A new wet storage facility would be located in one of two 60-plus ha (150-plus acres) undeveloped areas near the H- and P-areas, respectively. Predominant land use at both areas is managed timber land. Construction activities, including laydown areas, would disturb 2.8 ha (7 acres) of land. This represents less than 5 percent of the available space at either area. A new wet storage facility would occupy 3,800 m^2 (41,000 ft^2) of land and would move 18,000 m^3 (24,000 yd^3) of soil. Neither construction nor operation of a new wet storage facility at either area would significantly impact land use patterns on the Savannah River Site.

F.4.1.2.2.2 Socioeconomics

As discussed in Section F.3.2 the total capital cost of a new wet storage facility is estimated to be \$449 million. Construction activities are projected to take 4 years. Assuming that the capital cost is evenly distributed over this 4-year period, the annual expenditures would be about \$112.2 million. This represents approximately 9.4 percent of the estimated FY 1995 total expenditures for the Savannah River Site (1,198 million). The relative socioeconomic impact from annual construction expenditures on the region of influence would be small but positive. The annual operations costs of a new wet storage facility are estimated to be $23.3 million for receipt and handling and $3.5 million for storage. These costs represent about 1.9 percent and 0.3 percent of FY 1995 total expenditures for the Savannah River Site. The relative socioeconomic impact from annual operation expenditures on the region of influence would be small.

Direct employment associated with construction of a new wet storage facility is estimated to be 157 persons. The relative socioeconomic impact from direct construction employment on the region of influence would be small. In addition, when compared to the projected FY 1995 work force at the Savannah River Site of approximately 20,000 persons, the relative socioeconomic impact of this temporary increase in construction employment would be insignificant. Direct employment associated with operations of a new wet storage facility is estimated to be 30 persons. The relative socioeconomic impact of this increase in operations employment would be small to both the region of influence and the Savannah River Site.

F.4.1.2.2.3 Cultural Resources

Impacts to cultural resources would be the same as for new dry storage (Section F.4.1.2.1.3).

F.4.1.2.2.4 Aesthetic and Scenic Resources

Impacts to aesthetic and scenic resources would be the same as for new dry storage (Section F.4.1.2.1.4).

F.1.4.2.2.5 Geology

Impacts to geology would be the same as for new dry storage (Section F.4.1.2.1.5).

F.4.1.2.2.6 Air Quality

Nonradiological Emissions: Construction of a new wet storage facility would necessitate the clearing and grading of approximately 2.8 ha (7 acres) of land. In comparison, approximately 3.7 ha (9 acres) of land would be disturbed by new dry storage construction. Therefore, air quality impacts associated with wet storage construction would be bound by those associated with new dry storage construction, as presented in Table F-25.

Operations-related impacts associated with wet storage would be similar to those discussed under existing facilities.

Radiological Emissions: Incident-free airborne releases from a new wet storage facility would be limited to radioactive noble gases and some radioactive iodine which could be released from the stored fuel prior to canning. The airborne materials released to the building atmosphere during incident-free operations would be filtered by the building heating and ventilation system. Radioactive and nonradioactive effluent gases would be routed through double-banked high-efficiency particulate air filters prior to release to the environment through an exhaust air system. The high-efficiency particulate air filter would have a minimum efficiency of 99.97 percent for 0.3-micron diameter particulates and would allow in-place dioctyl phthalate testing.

The new wet storage facility would discharge all ventilated gas, except truck exhaust, to the facility's exhaust system. The truck exhaust would be discharged directly to the environment during cask off-loading operations in the truck receiving area. The exhaust air system would employ a detector to monitor ^{137}Cs. For other building areas which would be sources of airborne radioactive contamination, the heating, ventilation, and air conditioning system would be designed to maintain airflow from areas of low potential contamination into areas of higher potential contamination. These airborne effluents would be required to be below the radioactivity concentration guides listed in DOE Order 5480.1B for both onsite and offsite concentrations (DOE, 1989b).

Air emissions from the new wet storage facility are expected to be similar to the air emissions from the IFSF at the Idaho National Engineering Laboratory. The annual air emission for the IFSF was designed to result in ground-level concentrations of less than 0.003 percent of DOE Order 5480.1B limits for uncontrolled areas.

Radiological emissions from the operation of the new wet storage facility were calculated based on the methodology and assumptions used in Appendix F, Section F.6. The annual emission releases from the wet storage facility during the receipt and unloading, and storage are provided in Section F.6.6.1.

F.4.1.2.2.7 Water Resources

The annual water usage during construction and operation of a new wet storage facility is estimated to be about 1.9 million l (502,000 gal) and 2.7 million l (720,000 gal), respectively. With an annual average water usage of approximately 88,200 million l (23,300 million gal) for the Savannah River Site, these amounts represent an increase of less than 0.01 percent for both. Therefore, a new wet storage facility would have minimal impact on water resources at the Savannah River Site.

Best-management practices during construction would prevent sediment runoff or spills of fuels or chemicals. Therefore, construction activities should have no impact on water quality at the Savannah River Site. The impact on water quality during operations would also be negligible. Existing water treatment facilities at the Savannah River Site could accommodate any new domestic and process wastewater streams from a new wet storage facility. The expected total flow volumes at the Savannah River Site would still be well within the design capacities of treatment systems at the Savannah River Site.

F.4.1.2.2.11 Occupational and Public Health and Safety

Emission-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Savannah River Site would be attributed to emissions of radioactive material that could be carried by wind offsite. The general public would be too far from the locations where handling activities or storage take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the storage facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.6 of this appendix. Table F-30 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Savannah River Site. Integrated doses for the duration of a specific implementation period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-30 Annual Public Impacts for Receipt and Storage of Foreign Research Reactor Spent Nuclear Fuel at the Savannah River Site (Implementation Alternative 5 of Management Alternative 1)

Facility	MEI Dose (mrem/yr)	MEI Risk (LCF/yr)	Population Dose (person-rem/yr)	Population Risk (LCF/yr)
Receipt/Unloading at:				
• BNFP	0.00065	3.3×10^{-10}	0.0045	0.0000023
• New Wet Storage Facility	0.00011	5.5×10^{-11}	0.0057	0.0000028
Storage at:				
• BNFP	7.5×10^{-9}	3.8×10^{-15}	4.8×10^{-8}	2.4×10^{-11}
• New Wet Storage Facility	1.2×10^{-9}	6.0×10^{-16}	6.2×10^{-8}	3.1×10^{-11}

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask), transferring the spent nuclear fuel from one facility to another, or preparing the spent nuclear fuel for shipment offsite. Analysis option 1C involves the receipt of 644 shipments of foreign research reactor spent nuclear fuel into an existing wet storage facility (RBOF and/or L-Reactor disassembly basin) during Phase 1, the preparation of 161 transportation casks for shipment to a wet storage facility at the end of Phase 1, and the receipt of 193 shipments of foreign research reactor spent nuclear fuel directly from the ports into the new wet storage facility after Phase 1 operations. It was assumed that at the end of a 10-year period, the foreign research reactor spent nuclear fuel would have decayed sufficiently to be accommodated in larger capacity transportation casks, such as those currently used in the United States for commercial spent nuclear fuel. For the purpose of this analysis, the transportation casks used for intrasite shipping are assumed to have a capacity four times as large as the capacity of the transportation casks used for the marine transport of the foreign research reactor spent nuclear fuel to the United States. The assumptions and methodologies used to calculate the doses to a working crew associated with the handling activities of the foreign research reactor spent nuclear fuel are described in Section F.5 of this appendix.

Table F-31 presents the population dose that would be received by the members of the working crew and the associated risk if that working crew handled the total number of transportation casks at the Savannah River Site. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the administrative limits at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job to prevent further exposure. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. For this dose, the associated risk of incurring an LCF would be 2.6 percent.

Table F-31 Handling-Related Impacts to Workers at the Savannah River Site (Implementation Alternative 5 of Management Alternative 1)

	Worker Population Dose	*Worker Population Risk*
[illegible]	[illegible]	[illegible]

Table F-33 Annual Waste Generated from New Wet Storage at the Savannah River Site (Implementation Alternative 5 of Management Alternative 1)

Waste Form	Baseline Site Generation	Wet Storage Generation	Percent Increase
High-Level (m^3/yr)	127,400[a]	none	0 percent
Transuranic (m^3/yr)	760	none	0 percent
Solid Low-Level (m^3/yr)	19,750	16[b] 1[c]	0.08 percent 0.005 percent
Wastewater (l/yr)	690,000,000	1,590,000[b] 400,000[c]	0.23 percent 0.06 percent

[a] *Total inventory (m^3) at the Savannah River Site.*

[b] *During receipt and handling*

[c] *During storage*

storage facility. Therefore, the impact of this waste on the existing Savannah River Site waste management capacities would be minimal.

F.4.1.3 Accident Analysis

An evaluation of incident-free operations and hypothetical accidents at the Savannah River Site is presented here, based on the methodology presented in Appendix F, Section F.6. The evaluation assessed the possible radiation exposure to individuals and general population due to the release of radioactive materials. The analyses are based on the same operations carried out at the different potential storage locations and the same accidents at any of the sites evaluated.

The radiation doses to the following individuals and the general population are calculated for accident conditions at the spent nuclear fuel storage facility:

- ***Worker***: An individual located 100 m (330 ft) from the radioactive material release point. For elevated release, the worker dose was calculated at a point of maximum dose. The distance at which the maximum dose occurs is frequently greater than 100 m (330 ft) for elevated release. The direction to the worker was chosen as the direction to the maximally exposed sector. The dose to the worker is calculated for the 50th-percentile meteorological condition (DOE, 1992a).

- ***Maximally Exposed Offsite Individual (MEI):*** A theoretical individual living at the storage site boundary receiving the maximum exposure. The individual is assumed to be located in a direction downwind from the release point. The dose to the MEI is shown for the 95th-percentile meteorological condition.

- ***Nearest Public Access Individual (NPAI).*** An individual stranded on a highway or public access road near to the facility at the time of an accident. The distance to the NPAI was chosen as the distance to the nearest public access point; the direction was chosen as the direction to that point. The dose to the NPAI is shown for the 95th-percentile meteorological condition.

- ***General Population Within an 80 km (50 mi) Radius of the Facility:*** The dose calculations are performed for the direction downwind from the release point that results in highest dose to the public. The dose to the population is shown for the 95th-percentile meteorological condition.

The radiation dose to individuals and the public resulting from exposure to radioactive contamination was calculated using external (direct exposure), inhalation, and ingestion pathways. Dispersion in air from point of release was estimated with both 50th-percentile and 95th-percentile meteorological conditions. The 50th-percentile condition represents the median meteorological condition. The 95th-percentile condition is defined as that condition which is not exceeded more than 5 percent of the time, and is more conservative than the 50th-percentile condition.

The ingestion dose is calculated by considering that the individual and the public would consume contaminated food produced in the vicinity [up to 80 km (50 mi)] of the accident. This is conservative, and it is expected that continued consumption of contaminated food products by the public would be suspended after a protective action guideline is reached. In 1991, the U.S. Environmental Protection Agency recommended protective action guidelines in the range of one to five rem whole-body exposure (EPA, 1991). To ensure a consistent analytical basis, no reduction of exposure due to a protective action guideline was used in this analysis.

Accidents considered for detailed analysis are similar to those analyzed in the Programmatic SNF&INEL Final EIS. The selection of accidents was based on the following considerations:

- Accidents in the Programmatic SNF&INEL Final EIS were reviewed to select reasonably foreseeable accidents. They are: (1) criticality caused by human error during operation, equipment failure, or earthquake; (2) mechanical damage to foreign research reactor spent nuclear fuel during examination and preparation (cropping off the aluminum and nonfuel end of a spent nuclear fuel element); and (3) accident involving an impact by either an internal or an external initiator with and without an ensuing fire.

Six accident scenarios were evaluated at each storage location using identical source terms (estimated amounts of radioactive material released during postulated accidents). The wet pool accidents are assumed to be cutting into the fuel region or mechanical damage due to operator error, an accidental criticality, and an aircraft crash into the water pool facility. The dry storage accidents are assumed to be cutting into the fuel region or mechanical damage during examination work and handling in a dry cell, dropping of a fuel cask, and an aircraft crash with an ensuing fire.

Table F-34 presents frequency and consequences in terms of mrem or person-rem, of postulated accidents to the offsite MEI, NPAI, and offsite population for the 95th-percentile meteorological conditions using the assumptions and input values discussed above. The worker doses are calculated only for the 50th-percentile meteorology. This is an individual assumed to be 100 m (330 ft) downwind of the accident. DOE did not estimate the worker population dose.

Multiplying the frequency of each accident times its consequences at the Savannah River Site and converting the radiation doses to LCF yields the annual risks associated with each potential accident at the Savannah River Site. These annual risks are multiplied by the maximum duration of the policy at each site to obtain conservative estimates of risks for the entire program at the Savannah River Site. These risk estimates are presented in Table F-35.

Table F-36 presents the frequency and consequences of the accidents analyzed for the Savannah River Site for wet storage (Implementation Alternative 5 of Management Alternative 1). Multiplying the frequency of each accident times its consequences at the Savannah River Site and converting the radiation doses to LCF yields the annual risks associated with each potential accident at the Savannah River Site. These annual risks are multiplied by the maximum duration of this implementation alternative at the Savannah

Table F-34 Frequency and Consequences of Accidents at the Savannah River Site

	Frequency (per year)	Consequences			
		MEI (mrem)	NPAI (mrem)	Population (person-rem)	Worker (mrem)
Dry Storage Accidents - New					
• Spent Nuclear Fuel Assembly Breach	0.16	0.24	0.068	9.2	28
• Dropped Fuel Cask	0.0001	0.018	0.00034	0.55	0.28
• Aircraft Crash w\Fire	1×10^{-6}	40	0.29	1300	120
Wet Storage Accidents at RBOF					
• Spent Nuclear Fuel Assembly Breach	0.16	0.0070	0.00039	0.23	0.14
• Accidental Criticality	0.0031	130	44	4,800	16,000
• Aircraft Crash	1×10^{-6}	4.1	0.98	150	400
Wet Storage Accidents at L-Reactor Basin					
• Spent Nuclear Fuel Assembly Breach	0.16	0.0003	0.00007	[illegible]	[illegible]

Table F-35 Annual Risks of Accidents at the Savannah River Site

	Risks

River Site to obtain conservative estimates of risks at the Savannah River Site. Table F-37 presents the risk estimates from this implementation alternative.

Table F-37 Annual Risks of Accidents at the Savannah River Site (Implementation Alternative 5 of Management Alternative 1)

	Risks			
	MEI (LCF/yr)	NPAI (LCF/yr)	Population (LCF/yr)	Worker (LCF/yr)
Wet Storage Facility - New				
• Spent Nuclear Fuel Assembly Breach	5.5×10^{-10}	3.1×10^{-11}	0.000019	8.8×10^{-10}
• Accidental Criticality	2.7×10^{-8}	1.5×10^{-8}	0.00060	0.0000020
• Aircraft Crash	2.1×10^{-12}	4.9×10^{-13}	7.5×10^{-8}	1.6×10^{-10}
BNFP				
• Spent Nuclear Fuel Assembly Breach[a]	2.8×10^{-9}	8.0×10^{-11}	0.0000023	5.2×10^{-11}
• Accidental Criticality[a]	1.3×10^{-7}	1.2×10^{-7}	0.000070	9.2×10^{-8}
• Aircraft Crash[a]	4.6×10^{-10}	1.6×10^{-11}	1.2×10^{-8}	2.8×10^{-10}

[a] *Emissions would be released through a tall stack.*

F.4.1.3.1 Secondary Impact of Radiological Accidents at the Savannah River Site

In the event of an accidental release of radioactivity, there is a potential for impacts to land uses, cultural resources, water quality, ecology, national defense, and local economies (secondary impacts). For this analysis, secondary impacts of radiological accidents involving foreign research reactor spent nuclear fuel have been qualitatively assessed based on the results of the accident calculations presented in Section F.4.1.3. Radiological accidents that would result in doses to the MEI of less than the annual Federal radiological exposure limit for the public of 100 mrem (10 CFR Part 20) were considered to have no secondary impacts.

The MEI dose provides a measure of the air concentration and radionuclide deposition at the receptor location. As such, it can be used to express the level of contamination from a given radiological accident. In estimating the human health effects from radiological exposure (as presented in Section F.4.1.3), the MEI dose evaluates four pathways: (1) air immersion, (2) ground surface, (3) inhalation, and (4) ingestion. In estimating the environmental effects from radiological exposure, however, only the air immersion and ground surface pathways need be considered.

At the Savannah River Site, the radiological accident with the highest MEI dose is the accidental criticality at a wet storage facility (Table F-34). For this accident, the MEI dose would be 170 mrem. For the air immersion and ground surface pathways only, the dose would be 50 mrem, (Table F-115A) which is lower than the 100 mrem limit used in this analysis. Local contamination would be likely around the dry storage facility, but is expected to be contained entirely within the boundaries of the Savannah River Site. Cleanup activities should be small and any impacts to land uses, cultural resources, water quality, and ecology would be reversible. No impacts to national defense or local economies would be expected.

F.4.1.4 Cumulative Impacts at the Savannah River Site

This section presents the cumulative impacts of the proposed action, potential impacts of other major contemplated DOE actions, and other offsite (non-DOE) facility impacts at the Savannah River Site. A major portion of the presentation is based on information included in the Interim Management of Nuclear Materials Final EIS for the Savannah River Site, issued in October 1995 (DOE, 1995b). The cumulative impacts include those associated with the handling and dry storage of foreign research reactor spent nuclear fuel at the Savannah River Site and the following existing or major foreseeable activities proposed for the site:

- The operation of the Vogtle Electric Generating Plant located approximately 16 km (10 mi) south west of the center of the Savannah River Site.
- The implementation of the preferred scenario in the Interim Management of Nuclear Materials EIS (DOE, 1995b).
- Shipment of aluminum-based spent nuclear fuel to the Savannah River Site for storage and disposal discussed in Appendix C of the Programmatic SNF&INEL Final EIS (DOE, 1995g).
- Completion of the construction and operation of the Defense Waste Processing Facility (DOE, 1994g).
- Processing of F-Canyon plutonium solutions to metal (DOE, 1994a).
- Treatment and minimization of radioactive and hazardous wastes at the site as identified in the Savannah River Site Waste Management Final EIS (DOE, 1995f).
- Construction of an accelerator for

Table F-38 Cumulative Impacts at the Savannah River Site

Environmental Impact Parameter	FRR SNF Contribution	Current Activities[a]	Other Activities[b]	Cumulative Impact
Land Use (acres)	9	9,075[c]	3,975	13,059
Socioeconomics (persons)	190[d]/30[e]	(f)	11,000[d]/6,200[e]	11,190[d]/6,230[e]
Air Quality (nonradiological)	See Table F-38A	See Table F-38A	See Table F-38A	See Table F-38A

Table F-38A Estimated Maximum Nonradiological Cumulative Ground-Level Concentrations of Criteria and Toxic Pollutants at the Savannah River Site Boundary[a]

Pollutant	Averaging Time	Regulatory Standard ($\mu g/m^3$)	Cumulative Concentration ($\mu g/m^3$)[b]
Carbon Monoxide	1-hour	40,000	331.7 (0.83%)
	8-hour	10,000	52.3 (0.52%)
Nitrogen Oxides	Annual	100	19.5 (19.5%)
Sulfur Dioxide	3-hour	1,300	1,159 (89.1%)
	24-hour	365	248 (68.2%)
	Annual	80	17 (21.3%)
Gaseous Fluorides	12-hour	3.7	1.38 (37.3%)
	24-hour	2.9	0.58 (20%)
	1 week	1.6	0.56 (34.8%)
	1 month	0.8	0.066 (8.2%)
Nitric Acid	24-hour	125	9.8 (7.8%)

a *Concentrations represent: foreign research reactor spent nuclear fuel management, other DOE-owned spent nuclear fuel management, defense waste processing facility operations, consolidated incineration facility operation, stabilization of Pu-solutions, waste management activities, tritium supply and recycling, disposition of surplus highly enriched uranium, storage and disposition of weapons-usable fissile materials, and stockpile and stewardship management program activities*

b *Numbers in parentheses indicate the percentage of the regulatory standard*

F.4.1.6 Irreversible and Irretrievable Commitments of Resources

The irreversible and irretrievable commitment of resources resulting from the construction and operation of facilities for the receipt and storage of foreign research reactor spent nuclear fuel would involve materials that could not be recovered or recycled or that would be consumed or reduced to unrecoverable forms. The construction and operation of facilities for foreign research reactor spent nuclear fuel facilities at the Savannah River Site would consume irretrievable amounts of electrical energy, fuel, concrete, sand, and gravel. Other resources used in construction would probably not be recoverable. These would include finished steel, aluminum, copper, plastics, and lumber. Most of this material would be incorporated in foundations, structures, and machinery. Construction and operation of facilities for foreign research reactor spent nuclear fuel management would also require the withdrawal of water from surface- and groundwater sources, but most of this water would return to onsite streams or the Savannah River after use and treatment.

F.4.1.7 Mitigation Measures

Mitigation is addressed in general terms and describes typical measures that the Savannah River Site could implement. The analyses indicate that the environmental consequences attributable to foreign research wastewater discharges, source reduction of air emissions, and potential procurement of products manufactured from recycled materials. Since 1991, waste (all types) generated at the Savannah River Site has decreased, with the greatest reductions in hazardous and mixed wastes. These reductions are attributable primarily to material substitutions (DOE, 1995g).

All foreign research reactor spent nuclear fuel activities at the Savannah River Site would be subject to a pollution prevention program. Implementation of the program plan would minimize waste generated by these activities (DOE, 1995g).

Cultural Resources: A Programmatic Memorandum of Understanding, ratified on August 24, 1990, between the DOE Savannah River Operations Office, the South Carolina State Historic Preservation Office, and the Advisory Council on Historic Preservation is the instrument for the management of cultural resources at the Savannah River Site. DOE uses this memorandum to identify cultural resources and develop mitigation plans for affected resources in consultation with the State Historic Preservation Office.

DOE would comply with the terms of the memorandum in support of foreign research reactor spent nuclear fuel activities at the Savannah River Site. For example, DOE would survey sites prior to disturbance and reduce impacts to any potentially significant resources discovered through avoidance or removal. Any artifacts encountered would be protected from further disturbance and the elements until removed (DOE, 1995g).

DOE conducted an investigation of Native American concerns over religious rights in the Central Savannah River Valley in conjunction with studies in 1991 related to a New Production Reactor. During this study, three Native American groups expressed concern over sites and items of religious significance on the Savannah River Site. DOE has included these organizations on its environmental mailing list, solicits their comments on NEPA actions on the Savannah River Site, and sends them documents about the Savannah River Site environmental activities, including those related to foreign research reactor spent nuclear fuel management considerations. These Native American groups would be consulted on any actions that may follow subsequent site-specific environmental reviews (DOE, 1995g).

Geology: DOE expects that there would be no impacts to geologic resources at the Savannah River Site under any storage option evaluated. Potential soil erosion in areas of ground disturbance would be minimized through sound engineering practices such as implementing controls for storm water runoff (e.g., sediment barriers), slope stability (e.g., rip-rap placement), and wind erosion (e.g., covering soil stockpiles). Relandscaping would minimize soil loss after construction was completed. These measures would be included in a site-specific Storm Water Pollution Prevention Plan that the Savannah River Site would prepare prior to initiating any construction (DOE, 1995g).

Air Resources: DOE would meet applicable standards and permit limits for all radiological and nonradiological releases to the atmosphere. In addition, the Savannah River Site would follow the DOE policy of maintaining radiological emissions to levels "as low as reasonably achievable" (ALARA). ALARA is an approach to radiation protection to control or manage exposures (both individual and collective) and releases of radioactive material to the environment as low as social, technical, economic, practical, and public policy considerations permit. ALARA is not a dose limit, but rather a process that has as its objective the attainment of dose levels as far below the applicable limits as practicable (DOE, 1995g).

Water Resources: DOE would minimize the potential for adverse impacts on surface water during construction through the implementation of a storm water pollution prevention plan that details controls for erosion and sedimentation. The plan would also establish measures for prevention of spills of fuel and chemicals and for rapid containment and cleanup (DOE, 1994g).

DOE could minimize water usage during both construction and operation of facilities by instituting water

use), installing flow restrictors, and using self-closing hose nozzles (DOE, 1995g).

Ecological Resources: DOE does not anticipate any impact on wetlands on the Savannah River Site as a result of the spent nuclear fuel program. In any case, it is DOE and the Savannah River Site policy to achieve "no net loss" of wetlands. Pursuant to this goal, DOE has issued a guidance document disposition. If the Idaho National Engineering Laboratory is not the site to manage DOE-owned spent nuclear fuel, foreign research reactor spent nuclear fuel could be received and managed at the Idaho National Engineering Laboratory until the selected site(s) would be ready to receive the foreign research reactor spent nuclear fuel. The construction of new facilities for managing foreign research reactor spent nuclear fuel is estimated to take about 10 years; this period is referred to as Phase 1. The period following Phase 1 until ultimate disposition is referred to as Phase 2 (approximately 30 years). The amount of spent nuclear fuel that could be received at the Idaho National Engineering Laboratory under the basic implementation of Management Alternative 1 is dictated by the distribution considered in the Programmatic SNF&INEL Final EIS. Accordingly, the Idaho National Engineering Laboratory could receive one-half of the foreign research reactor spent nuclear fuel under the Decentralization and the 1992/1993 Planning Basis alternatives, all of the TRIGA-type foreign research reactor spent nuclear fuel under the Regionalization by Fuel Type alternative, only the foreign research reactor spent nuclear fuel from Western ports under the Regionalization by Geography Alternative, or all foreign research reactor spent nuclear fuel under the Centralization Alternative. As discussed in Section 2.6.4.1, the split of foreign research reactor spent nuclear fuel evenly between the Savannah River Site and the Idaho National Engineering Laboratory under the Decentralization and 1992/1993 Planning Basis alternatives in the Programmatic SNF&INEL Final EIS was not considered to have a practical basis, and was therefore not evaluated in detail.

As a potential Phase 1 site, the Idaho National Engineering Laboratory would receive and manage foreign research reactor spent nuclear fuel at existing dry and wet storage facilities. The existing facilities identified for this purpose would be the FAST facility in CPP-666, the IFSF in CPP-603, and the CPP-749 storage area. Descriptions of these facilities are provided in Appendix F, Section F.3.

The FAST facility is a modern underwater storage facility which has been used in the past for receipt and storage of foreign research reactor spent nuclear fuel. It has the capability to receive and unload spent nuclear fuel casks at a rate of approximately five per week. Storage capacity for up to 8,400 foreign research reactor spent nuclear fuel elements could be provided in a 10-year period by using the spent nuclear fuel storage racks that would be installed. The capability of the FAST facility to receive foreign research reactor spent nuclear fuel in the near term is limited due to the number of activities scheduled through FY 1998. Considering these activities, DOE estimates that 3,600 elements could be received by the end of 1999 at the FAST facility.

The IFSF is a shielded dry storage vault originally constructed for Fort St. Vrain reactor fuel. The storage capacity available is for approximately [illegible] foreign research reactor spent nuclear fuel elements of foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory was shown earlier in Figure F-18.

An additional option to enhance storage capacity during Phase 1 would be to use the existing facilities to unload the transportation casks, and provide storage capacity in dry storage casks which would be placed near the existing facility.

As a Phase 2 site, the Idaho National Engineering Laboratory would continue to receive and manage foreign research reactor spent nuclear fuel at existing facilities until a new dry storage facility becomes operational at the site. Foreign research reactor spent nuclear fuel managed at existing facilities would then be transferred to the new facility where it would remain until ultimate disposition. The new facility would also receive foreign research reactor spent nuclear fuel shipments directly from ports after Phase 1 concluded. Dry storage encompasses both the dry vault design and the dry cask design as described in Section 2.6.5.1.1.

The analysis of environmental impacts from management of foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory is based on the above considerations. The analysis options selected do not represent all possible combinations, but a reasonable set which provides a typical, and in many cases, bounding estimate of the resulting impacts.

The specific analysis options under the basic implementation of Management Alternative 1 are as follows:

2A. The Idaho National Engineering Laboratory would receive foreign research reactor spent nuclear fuel during Phase 1 and manage it at the FAST, the IFSF, and/or the CPP-749 facilities. For the purpose of this analysis, the amount of fuel to be stored is all foreign research reactor spent nuclear fuel that would be received in a 10-year period (17,500 elements). The fuel would be shipped offsite at the end of Phase 1.

2B. Foreign research reactor spent nuclear fuel managed under analysis option 2A would be transferred to a newly constructed dry storage facility where it would be managed until ultimate disposition. Spent nuclear fuel arriving at the United States after Phase 1 concludes would be received and managed at the new dry storage facility until ultimate disposition. For the purpose of this analysis, the amount of spent nuclear fuel that would be stored in the dry storage facility would be all the foreign research reactor spent nuclear fuel eligible under

- Under Implementation Subalternative 1b (Section 2.2.2.1), the Idaho National Engineering Laboratory would receive only HEU from the reactors eligible under the policy. The amount of HEU would be approximately 4.6 MTHM, representing 11,200 elements. The impacts from the storage of this amount of fuel at the Idaho National Engineering Laboratory would be bounded by analysis options 2A and 2B above.

- Under Implementation Subalternative 1c (Section 2.2.2.1), the Idaho National Engineering Laboratory would receive target material in addition to the foreign research reactor spent nuclear fuel considered under the basic implementation of Management Alternative 1. The

- Under Implementation Alternative 6 (Section 2.2.2.6), DOE and the Department of State would consider chemical separation of foreign research reactor spent nuclear fuel in the United States. As noted in the discussion in Section 2.3.6, chemical separation of both aluminum-based and TRIGA foreign research reactor spent nuclear fuel is evaluated for the Idaho National Engineering Laboratory.

Under Management Alternative 3 (Hybrid Alternative), as discussed in Section 2.4, the Idaho National Engineering Laboratory would receive the foreign research reactor TRIGA spent nuclear fuel. This spent nuclear fuel would be managed at the Idaho National Engineering Laboratory in existing facilities until ultimate disposition. The amount of TRIGA spent nuclear fuel that would be stored is 4,900 elements, 1.0 MTHM, 19 m^3 (670 ft^3).

F.4.2.1 Existing Facilities (Phase 1)

Analysis option 2A utilizes existing facilities for receipt and storage of foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory. The impacts from this analysis option include only those related to operations, specifically: socioeconomics, occupational and public health and safety, utilities and energy, air quality, and waste management. For this analysis, it was assumed that the amount of foreign research reactor spent nuclear fuel to be received at this management site is the maximum and the receipt rate is uniform at approximately 1,800 elements per year.

F.4.2.1.1 Socioeconomics

Potential socioeconomic impacts associated with analysis option 2A would be attributable to staffing requirements at existing facilities (FAST and IFSF). Currently, these facilities are being used to store spent nuclear fuel, so any incremental staffing requirements related to foreign research reactor spent nuclear fuel storage would be insignificant. All personnel required for the operation and support of the existing facilities could be acquired from the current work force at the Idaho National Engineering Laboratory. Use of the current work force would not result in any net socioeconomic impact relative to baseline environmental conditions. In fact, using the current work force would partially compensate for the decline in employment expected from changes in site mission.

F.4.2.1.2 Occupational and Public Health and Safety

Emission-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory would be attributed to emissions of radioactive material that could be carried by wind offsite. The public would be too far from the locations where handling activities or storage would take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the storage facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.6 of this appendix. Table F-39 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Idaho National Engineering Laboratory. Integrated doses for the duration of a specific period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-39 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage in Existing Facilities at the Idaho National Engineering Laboratory (Phase 1)

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• IFSF/CPP-749 (dry storage)	0.00056	2.8×10^{-10}	0.0045	0.0000023
• FAST (wet storage)	0.00038	1.9×10^{-10}	0.0031	0.0000016
Storage at:				
• IFSF/CPP-749 (dry storage)	0	0	0	0
• FAST (wet storage)	3.8×10^{-9}	1.9×10^{-15}	3.1×10^{-8}	1.6×10^{-11}

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask), transferring the spent nuclear fuel from one facility to another, or preparing the spent nuclear fuel for shipment offsite. Analysis option 2A involves the receipt of 644 shipments of foreign research reactor spent nuclear fuel into existing storage facilities (IFSF/CPP-749 and FAST) during Phase 1, and the preparation of 161 transportation casks for shipment at the end of Phase 1. It was assumed that at the end of a 10-year period, the foreign research reactor spent [illegible]

Table F-40 Annual Utility and Energy Requirements for Foreign Research Reactor Spent Nuclear Fuel Storage at Existing Facilities at the Idaho National Engineering Laboratory (Phase 1)

Commodity	Baseline Site Usage	FAST	IFSF	Percent Increase
Electricity (MW-hr/year)	208,000	1,490	1,490	0.72 percent
Water (l/year)	6,500,000,000	1.93 million	2.12 million	0.033 percent
Fuel (l/year)	11,123,400	0	0	0 percent

The requirements for all storage options represent a small percentage of current requirements. No new generation or treatment facilities would be necessary. Increases in the Idaho National Engineering Laboratory fuel consumption would be minimal because overall activity would not increase due to changes in the Idaho National Engineering Laboratory mission and the general reduction in employment levels. The overall impacts of any of the storage options at the Idaho National Engineering Laboratory on materials, utilities, and energy resources would be minimal.

The existing capacities and distribution systems at the Idaho National Engineering Laboratory for electricity, steam, water, and domestic wastewater treatment are adequate to support the receipt and storage of foreign research reactor spent nuclear fuel for all storage options.

Some of the electric power at the Idaho National Engineering Laboratory is generated onsite, and the remainder is provided by the Idaho Power Company. The Utah Power and Light Company Antelope Substation, which is located on the Idaho National Engineering Laboratory, connects to the Scoville Substation, from which electricity is distributed to various facilities over a 138-kilovolt loop at the Idaho National Engineering Laboratory.

All water supplies for the Idaho National Engineering Laboratory are obtained from the Snake River Plain aquifer through wells. Pumping totals approximately 7 million m^3 per year (1.8 billion gallons per year). ICPP has a coal-fired steam system. Natural gas is not used at the Idaho National Engineering Laboratory.

F.4.2.1.4 Waste Management

Waste production associated with the operation of the FAST and IFSF facilities is characteristic to wet and dry storage, respectively, and is discussed in detail in Sections F.4.2.2.1.13 and F.4.2.2.2.13.

F.4.2.1.5 Air Quality

Nonradiological Emissions: It is expected that the ambient concentration levels from incident-free operation of existing facilities would not change from baseline concentrations due to the addition of foreign research reactor spent nuclear fuel. The baseline ambient concentrations are given in Table F-41. They are all below applicable standards and guidelines.

Radiological Emissions: Radiological emissions from the receipt and storage of foreign research reactor spent nuclear fuel in the existing facilities at the Idaho National Engineering Laboratory are discussed in Section F.4.2.1.2.

F.4.2.1.6 Water Resources

The use of FAST and ISFS facilities for the interim storage of foreign research reactor spent nuclear fuel would not change the current levels of water and usage of these facilities. Nor would it change thermal discharges from cooling water or the quantity or quality of radioactive and nonradioactive wastewater effluents.

Table F-41 Maximum Impacts to Nonradiological Air Quality from Spent Nuclear Fuel[a,b] at Existing Facilities at the Idaho National Engineering Laboratory (Phase 1)

Pollutant	Averaging Time	Applicable Standard ($\mu g/m^3$)[c]	Maximum Baseline Concentration ($\mu g/m^3$)	Baseline plus Foreign Research Reactor Spent Nuclear Fuel ($\mu g/m^3$)	Percent of Standard
Criteria pollutants					
• Carbon Monoxide	1-hr	40,000	1,200	1,200	3.8
• Nitrogen Dioxide	Annual	100	14.1	14.1	14.1
• Lead	Quarterly	1.5	0.002	0.002	0.1
• Particulate Matter (PM_{10})	24-hr Annual	150 50	112 19	112 19	75 38
• Sulfur dioxide	3-hr 24-hr Annual	1,300 365 80	534 238 4.2	534 238 4.2	41.1 65.3 5.3
Other pollutants mandated by Idaho					
• Total Suspended Particulates	24-hr Annual	150 60	120[d] 45	120 45	80 75
• Fluorides	Monthly Bimonthly Annual	62,168 46,626 31,084	0 0 0	0 0 0	0 0 0
Hazardous/toxic air pollutants (carcinogens)					
• Ammonia Hydroxide	8-hr	180	0.33	36	20
• Benzene	Annual	12	0.029	0.029	16
• Formaldehyde	Annual	770	0.012	0.012	16
• Hexone	8-hr	2,100	0	0	0
• Hydrofluoric Acid	8-hr	25	0	0	0
• Tributylphosphate	8-hr	25	0	0	0

a *Source: (DOE, 1995g).*

b *Listed concentrations are the maximum of those calculated at the Idaho National Engineering Laboratory site boundary, public access roads inside the Idaho National Engineering Laboratory site boundary, and the Craters of the Moon National Monument.*

c *To convert to $\mu g/ft^3$, multiply by 0.0283.*

d *The background concentration for the 24-hour standard is the same as the background for annual average concentration.*

Interim storage of foreign research reactor spent nuclear fuel in existing facilities would not affect the quality of water resources because it would be stored in contained storage pools or above-grade and below-grade dry storage containers isolated from the environment.

With respect to accident conditions, the Programmatic SNF&INEL Final EIS concluded that on the basis of a bounding accident scenario for high-level waste tank failure, accidental leakage would cause negligible impacts to water resources (DOE, 1995g).

F.4.2.2 New Facilities (Phase 2)

Analysis options 2B and 2C involve the use of new facilities. The environmental impacts analyzed relate to the construction and operation of these new facilities. The impacts include: land use; socioeconomics; cultural resources; aesthetic and scenic resources; geology; air and water quality; ecology; noise; traffic and transportation; occupational and public health and safety; materials, utilities and energy; and waste management.

The impacts are presented in terms of storage technologies: dry storage in Sections F.4.2.2.1 and wet storage in Section F.4.2.2.2. Accident analysis, which is associated primarily with the storage technology rather than specific facilities, is presented in Section F.4.2.3.

F.4.2.2.1 Dry Storage

Analysis option 2B is associated with dry storage of foreign research reactor spent nuclear fuel in new facilities. This analysis option would require the construction of a new dry storage facility at the Idaho National Engineering Laboratory. The dry storage option encompasses both the dry vault design and the dry cask design as described in Section 2.6.5 of this EIS and earlier in this appendix. There are no environmental impact parameters that would discriminate between the two designs. For the purpose of this analysis, the impacts from the larger dry vault design are presented.

F.4.2.2.1.1 Land Use

A new dry storage facility could be located in one of several developed areas, including the ICPP. These areas, which have already been developed for industrial use, occupy about 4,560 ha (11,400 acres). Construction activities, including laydown areas, would disturb 3.7 ha (9 acres) of land. This represents about 0.06 percent of the developed space at these areas. A new dry storage facility would occupy 5,000 m^2 (54,000 ft^2) of land and would move 11,000 m^3 (14,400 yd^3) of soil. Neither construction nor operation of a new dry storage facility at any of the areas would significantly impact land use patterns on the Idaho National Engineering Laboratory.

F.4.2.2.1.2 Socioeconomics

As discussed in Section F.3.1.1 the total capital cost of a new dry storage facility is estimated to be $370 million. Construction activities are projected to take 4 years. Assuming that the capital cost is evenly distributed over this 4-year period, the annual expenditures would be about $92.5 million. This represents approximately 15.4 percent of the estimated FY 1995 total expenditures for the Idaho National Engineering Laboratory (600 million). The relative socioeconomic impact from annual construction expenditures on the region of influence would be positive. The annual operations costs of a new dry storage facility are estimated to be $15.6 million for receipt and handling and $0.6 million for st