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VOL. 3 OF 5

Alternatives For Managing Wastes From Reactors And Post-Fission Operations In The LWR Fuel Cycle

MASTER

**Volume 3:
Alternatives For Interim Storage And Transportation**

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Alternatives For Managing Wastes From Reactors And Post-Fission Operations In The LWR Fuel Cycle

MAY 1976

United States
Energy Research & Development
Administration

**Volume 3
(3 of 5 Volumes)**

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**Report Coordinated By
Battelle, Pacific Northwest Laboratories
At The Request Of The
Division Of Nuclear Fuel Cycle And Production**

FOREWORD

This report describes technical alternatives for managing wastes from the back end of the commercial LWR fuel cycle. It describes waste types, the alternative technologies that can be used to manage the wastes, and the state of availability of those technologies.

The report does not select preferred waste management technologies or make comparative assessments. Its purpose is to provide a comprehensive digest of technical information expected to serve as a basis for future decisions, assessments, and environmental impact statements. The technology is still evolving, and this report will be subjected to public review and comment. Its contents therefore serve as a baseline subject to future supplementation and updating.

There are four major waste management functions: treatment, interim storage, transportation, and final storage or disposal. This report is structured in five volumes that reflect these functions. Volume 1 summarizes the report, describes alternatives for the back end of the LWR cycle, and provides background information on waste types and quantities. Volume 2 describes alternatives for waste treatment. Volume 3 discusses interim storage and transportation. Alternatives for final storage and disposal are described in Volume 4. The Appendices, Volume 5, provide supplementary information, with emphasis on characteristics of geologic formations that might be used for final storage or disposal.

Preparation of this report was started in August of 1975, in response to a directive from ERDA's Nuclear Fuel Cycle and Production Division. The report was authored by personnel of ERDA's contractors, of industry, and of universities; coordination was provided by Battelle's Pacific Northwest Laboratories. The report was completed in March of 1976. The cutoff date for characterizing the status of the technologies was September 1, 1975.

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INTRODUCTION

Alternative methods for interim storage and transportation of radioactive wastes are discussed in this volume. Storage operations are considered interim when a) monitoring and human control are provided and b) subsequent action involving treatment, transportation, or disposal is expected. Examples of interim storage technologies are:

- spent fuel storage in water basins
- tank storage of liquids
- canister storage in air-cooled vaults.

Interim-stored wastes are awaiting treatment or final storage or disposal in the form resulting from the alternative treatments presented in Volume 2.

Interim storage provides one or more of several waste management functions. These include:

- development and maintenance of a materials backlog
- holdup during decay of short-lived radioactivity
- accumulation of inventory needed for economic treatment or transportation
- holdup prior to final storage or disposal.

Spent fuel storage, discussed in Section 17, provides examples of most of these functions for interim storage. Spent fuel may be considered a waste or an energy resource subject to reprocessing. As a waste, spent fuel would be held in interim storage until transported for final storage or disposal. During interim storage, radiation and heat emissions would decrease as a result of decay of short-lived radioactive nuclides. If the spent fuel is to be reprocessed, interim storage would be utilized to permit short-lived radionuclides to decay for easier handling; also a backlog of spent fuel will be maintained to assure supply for the reprocessing plant.

Reuse is a special case discussed in the Interim Storage of Chop-Leach Fuel Bundle Residues, Section 18. If reuse is implemented, the interim storage capacity required for this waste would be reduced.

Sections 19 and 21 address interim storage of liquid and solidified high-level waste, respectively. Liquid high-level waste is stored in tanks for a period of up to 5 years to reduce radioactivity and heat emission through radioactive decay. Interim storage of solidified high-level waste would be used prior to final storage or disposal and for any wastes retrieved from a final storage repository.

Interim storage of solid and immobilized non-high-level wastes is considered in Section 20. Section 22 presents information on transportation of radioactive wastes and spent fuel elements.

17.0 INTERIM STORAGE OF SPENT FUEL ELEMENTS

17.0 INTERIM STORAGE OF SPENT FUEL ELEMENTS17.1 SUMMARY

In the uranium fuel cycle initially conceived for light water reactors (LWR), spent fuel was to be discharged from the reactors and then allowed to cool (at the reactor sites) for five months. This cooling period would allow short-lived radioactive isotopes to decay and thereby reduce the heat generated. After the cooling period, the fuel was to be shipped to a chemical reprocessing plant, which would recover uranium and plutonium. Accordingly, existing storage facilities at LWR sites were designed for only short-term storage of spent fuel. More recently, however, spent fuel has been recognized as a possible waste form⁽¹⁾ suitable for interim storage and even ultimate disposal.*

Alternatives for managing spent fuel include:

- Storage of unpackaged fuel in water-cooled basins or air-cooled vaults where reprocessing is not expected to be delayed more than 10 to 20 years.
- Storage of packaged spent fuel in water-cooled basins, air-cooled vaults, concrete surface silos, geologic formations, or near-surface heat sinks where reprocessing is expected to be delayed beyond 10 to 20 years.

Table 1 compares the storage alternatives. Alternatives not considered include fuel dissolution and subsequent solidification without separation of uranium and plutonium.

The fuel cladding is the initial barrier for confinement of fuel core material for all modes of spent fuel storage. The cladding withstands a far more-severe environment in the reactor than it would encounter in a storage facility although for a shorter period of time. In concept, a storage facility is designed to protect the fuel cladding against mechanical, chemical, or thermal damage; the storage facility provides for a safe, subcritical arrangement of fuel assemblies and adequate shielding under normal operating conditions or during extreme natural phenomena.⁽²⁾ Additional confinement is provided by cleanup of the effluent coolant streams when bare fuel is stored, and/or by the container if the fuel has been packaged.

Water-cooled storage of either bare spent fuel or packaged fuel would normally be in a near-surface facility with water-filled cells of heavily reinforced concrete lined with stainless steel. Auxiliary equipment would cool the water and maintain water quality. The technology of water basin

* Whether or not respect for the principle of conserving energy resources would require the U.S. government to reprocess spent fuel, even though the original owner had declared it to be waste, is beyond the scope of this report.

TABLE 1. Summary Comparison of Spent Fuel Storage Alternatives

<u>Storage Alternative</u>	<u>Confinement Barriers in Addition to Cladding</u>	<u>Means of Heat Removal</u>	<u>Method of Controlling Fuel Cladding Corrosion</u>	<u>Maintenance Requirements</u>	<u>Land Use (Surface)</u>
Unpacked Storage Water basin	Filters	Forced circulation of basin water	Water quality control	High	Low
Air-cooled vault	Filters	Forced circulation of air	Low temperature	Moderate	Moderate
Packaged Storage Water basin	Filters & package	Forced circulation of basin water	Packaged in inert or noncorrosive medium	High	Low
Air-cooled vault	Package	Natural circulation of air	Packaged in inert or noncorrosive medium	Low	Moderate
Concrete surface silo	Package	Natural circulation of or conduction to air	Packaged in inert or noncorrosive medium	Low	High ^(a)
Geologic	Package, hole liner & filters	Conduction to earth	Packaged in inert or noncorrosive medium	Moderate	Low
Near-surface heat sink	Package, hole liner	Conduction to earth	Packaged in inert or noncorrosive medium	Low	High

a. Estimated to be ~120,000 m² per 1500 MT, assuming one MT per storage unit.

storage of bare spent fuel is well developed and has been proven by about 30 years of operating experience (sequential storage) at reactor sites and fuel reprocessing plants.

Near-surface storage with forced draft air cooling is feasible for unpackaged LWR fuel that has been out of the reactor at least 3 to 4 years. This type facility would consist of reinforced Class I concrete storage vaults with auxiliary equipment to circulate and filter the coolant air. The fuel would be stored vertically in sleeves supported by top and bottom tube sheets, which would form inlet and discharge plenums to distribute and collect the air.

Two basic concepts using natural draft cooling of spent fuel have been proposed. In these concepts, the decay heat from residual fission products in the fuel creates a natural draft sufficient to cool the fuel. However, since the pressure of natural-draft air would be insufficient to force the air through an effective filtration system, packaging of the fuel would probably be required to reduce the probability of core material becoming airborne.

One natural-draft cooling concept calls for a large concrete structure similar to the storage vault for forced-draft air cooling. Packaged fuel is suspended in metal sleeves. Cooling air enters the sides of the vault and passes up through and out the top of the sleeves to the exhaust stack. In a second concept, spent fuel assemblies (possibly packaged assemblies) are sealed in thick low-carbon steel containers and placed in large cylindrical concrete housings (silos) located outdoors. Fuel is cooled either by convection of air between the steel container and the concrete silo or by conduction through the concrete to the atmosphere. The cooling is completely passive in these concepts and requires little maintenance and only minimal surveillance. Container failure is detected by an area monitoring system. Failed containers are removed from storage for repair and are returned to storage.

Two passive cooling concepts that use the earth as a heat sink have been tested. In one concept, packaged fuel is stored in holes in a mined bedded salt deposit (or other geologic formation). In the second concept, fuel packages are stored in holes near the earth's surface. In both concepts, concrete housings (hole liners) protect fuel packages against corrosion.

Some packaging of fuel assemblies (sealing of assemblies in a container with a heat transfer medium) would be required to seal leaking fuel assemblies and thus minimize radioactive releases. When reprocessing of the fuel is delayed, packaging of the fuel assembly would increase cladding integrity during long-term storage.

Spent fuel will probably not be packaged for storage until discharged fuel has cooled at least 3 to 4 years to decrease heat generation and

Process Summary

Storage Description. Near-surface, water-filled cell of heavily reinforced concrete lined with stainless steel.

Development Status. Demonstrated by about 30 years experience at reactor sites and fuel reprocessing plants.

Heat Removal. Basin water circulation with rejection to atmosphere.

Materials Used. Concrete and stainless steel.

Confinement Barriers. Cladding, basin water, and carbon and HEPA filters.

Maintenance Required. Filtration systems, heat removal systems, transport systems, and radiation monitoring systems.

Waste Sources. Radioactivity collected by filter systems and secondary solid waste from facility operation and maintenance.

Primary Advantages. Low fuel temperatures, additional confinement barriers, the large heat sink and transparent radiation shield provided by the water.

Primary Disadvantages. Requirement for stainless steel basin liner, water quality control, and substantial water supply.

radiation. The fuel would likely be packaged at central storage facilities rather than at reactor sites, according to studies⁽³⁾ that show the economic advantages of fuel reprocessing in central processing stations. Also, shipping cask volumes and therefore shipping cask weights can be reduced by shipping unpackaged rather than packaged assemblies.

17.2 STORAGE OF UNPACKAGED SPENT FUEL

17.2.1 Storage in Water-Cooled Basins

17.2.1.1 Summary

A facility for water-cooled storage of unpackaged (as-discharged) spent fuel is shown on the process summary sheet and would consist of near-surface, water-filled cells of heavily reinforced concrete lined with stainless steel and having auxiliary equipment to cool and control the quality of the water. In addition to serving as a heat transfer medium, water provides radiation shielding, partial containment for escaping radioactivity, and a heat sink under some accident conditions. Also, water allows visual observation of the stored fuel. The technology of water basin storage of bare spent fuel is well developed and demonstrated by about 30 years of experience (sequential storage) at reactor sites and fuel reprocessing plants.

17.2.1.2 Facility Description

Spent fuel from LWR's is currently stored in water-cooled basins. Most of these basins are at reactor sites and are designed to hold from 50 to 150 MT of irradiated fuel.⁽⁴⁾ A recently designed facility at a reprocessing plant has a capacity for 400 MT of spent fuel.⁽⁵⁾

A concept for a separate facility (i.e., not located at a reactor or fuel reprocessing site) for storage of spent fuel suggests the use of modular basins with each basin having a capacity for about 500 MT of spent fuel. The facility (Figure 1) is designed to receive, handle, decontaminate, and reship spent fuel casks; to remove irradiated fuel from casks; and to place the fuel in a storage basin. The basin, constructed of heavily reinforced concrete and lined with stainless steel, provides biological shielding, protection from natural phenomena, and secondary confinement of radioactivity. Fuel assemblies are stored in underwater racks and are so spaced as to prevent formation of critical arrays. A process building covers the basins to protect operating personnel, cranes, and other support equipment from the weather.

FIGURE 1. Block Diagram of BFRSS Operation. From Barnwell FRSS Application. Schematic of fuel receiving and storage operations (from Barnwell FRSS Operation)⁽⁵⁾

17.2.1.3 Heat Removal

Decay heat is transferred from the spent fuel to the basin water. The temperature of the basin water is usually maintained below 50°C by circulating part of the water through heat exchangers and back to the basin.* Cooling water for the heat exchangers discharges the heat to a cooling tower or a cooling pond. Storage basins of some of the more-recently licensed reactors have a cooling capacity of about 8 MW.⁽⁴⁾ Long-term storage facilities could be designed to provide additional cooling capacity as the amount of stored fuel (and the number of storage modules) increases. Typically, a storage facility sized to receive 1500 MT per year of fuel with various ages (equivalent to average age of one year) would have a maximum decay heat load of 15 MW.⁽⁵⁾ Thereafter, decay would reduce the power of stored fuel as fast as new fuel is added.

17.2.1.4 Criticality Control

Critically-safe storage of spent fuel could be assured by using storage racks designed to maintain subcritical spacing.^(2,6) Properly designed racks would remain stable even under conditions encountered during extreme natural phenomena.⁽²⁾ Enclosing each fuel assembly in a neutron-absorbing stainless steel structure has been proposed as a method of increasing the packing density in the basins without compromising nuclear safety. Figure 2 compares the present open design with a commercially available enclosed design.⁽⁷⁾

* The use of a closed cycle for the basin water reduces the potential for release of radioactivity to the environment.

FIGURE 2. Water Basin Storage with Standard and High-Capacity Rack Design⁽⁷⁾

17.2.1.5 Corrosion

Irradiated fuel is currently stored in water-filled basins, but the average storage time has been relatively short (up to ten years). The presence of halide ions and caustic in the water can cause pitting, stress corrosion cracking, and intergranular corrosion of the kinds of stainless steel cladding found on some commercial fuels. Halide ions can also cause stress corrosion cracking and accelerated uniform corrosion of the zirconium alloy cladding used on other commercial fuels. Corrosion by halide or other ions is minimized by using high purity deionized water in the basins. Corrosion rates of both stainless steel and zirconium alloys have been extremely low during the relatively short storage time of 10 years.

The effects of corrosion after extremely long storage times, on the order of 100 years, may need to be examined further to determine whether accelerated corrosion, microstructural changes, or changes in mechanical properties occur in long storage times at relatively low temperatures. Effects of long-term water storage that have not been fully explored are stress corrosion cracking and intergranular corrosion of stainless steels, and hydrogen absorption and precipitation by the zirconium alloys.

17.2.1.6 Radioactive Monitoring and Confinement

During operation of the storage facility, radioactive materials (radio-nuclides) could be released during receiving, handling, and storage of LWR fuel assemblies whose cladding has been breached during irradiation or subsequent handling. Fuel assemblies in which the cladding has failed, either before receipt at the storage facility or during storage, would be sealed in a container (packaged, Section 17.3) to confine the radioactivity. Radio-nuclides present in fuel cores include unreacted uranium, transuranium and other nuclides produced by nuclear reactions within the fuel, fission products, and nuclides produced by activation of fuel assembly structural materials. Small amounts of this radioactivity could contaminate coolant, ventilation air, off-gas, and solid wastes that leave the facility. Monitoring on a continuous basis and treatment when necessary would minimize the quantity of radionuclides released to the surroundings and to the ultimate disposal facility.

Basin water circulation in closed-loop systems through a filter-demineralizer system maintains water purity. This system filters small amounts of radioactive contaminants that have escaped from the fuel, and it removes materials from the water that could cause fuel cladding corrosion. The filter-demineralizer system is designed to be effective at temperatures well above normal operation. Periodically the system is backwashed and rejuvenated with water and chemicals that are discharged to a liquid-waste disposal system.

The basin ventilation system maintains the concentrations of radioactive gaseous contaminants at as low a level as practical.^(2,6) Both HEPA and carbon filters (or silver zeolite absorbers) can remove particulates and iodine from ventilating air. Demisters may be required to remove entrained water from the air. Basins at a fuel reprocessing site can share gas processing systems with the reprocessing plant. Treatment of off-gases to remove tritium and noble gases would be more feasible with this facility than with most others because of the low volume of air vented.

The facility buildings would be equipped with radiation monitoring devices around the fuel handling equipment and with continuous off-gas monitors in the basin area⁽⁸⁾ to detect and report abnormal radioactivity.

The design of water-cooled basins includes engineered safety features and automated protective systems and dictates operating procedures. For all practical purposes, the design minimizes the probability of a serious accident that would release radioactive contaminants.

17.2.1.7 Maintenance

Maintenance practices and procedures for fuel storage facilities are well established. Water treatment and air ventilation systems require routine maintenance to ensure proper operation of pumps and fans. Solid contaminated wastes (Section 3.0) from basin operation, including demineralizer resins, filters, tools, protective clothing, and miscellaneous trash could be compacted and packed in drums for shipping to a land-burial site.

17.2.1.8 Status of Technology

17.2.1.8.1 Current. The technology of water basin storage of spent fuel is well developed and demonstrated by approximately 30 years experience at reactor sites and fuel reprocessing plants. In new installations, the fuel assembly spacing probably would be determined by economics, and the use of high-capacity neutron absorbing racks would be considered.

17.2.1.8.2 Estimated Time of Readiness for Practical Design. Design and construction capabilities are available for water-cooled basin storage of spent fuel at reactor sites, at fuel reprocessing sites, or at separate locations. The estimated time required to build, license, and begin operating a storage facility is 6 to 8 years.

17.2.1.8.3 Major Development. Development work that would enhance the existing technology of water basin storage (as well as other storage concepts), though not required for design, include:

- Development of improved methods to detect and isolate failed fuel.
- Development of improved instrumentation to determine fissile content of fuels.
- Development of improved methods to predict stability of spent fuel during transport and storage.

Process Summary

AIR-COOLED STORAGE — UNPACKAGED SPENT FUEL

Storage Description. Near-surface, heavily reinforced concrete structure (Class 1).

Development Status. The basic technology is demonstrated by an existing facility for air-cooled storage of packaged *unsealed) assemblies from gas-cooled reactors.

Heat Removal. Forced air circulation.

Materials Used. Concrete and low carbon steel.

Confinement Barriers. Cladding and filters (HEPA and carbon).

Maintenance Required. Forced draft system including power supply and filters (HEPA and carbon), radiation monitoring system, and fuel transport equipment.

Waste Sources. Radioactivity collected by the filter system and secondary solid wastes from facility operation and maintenance.

Primary Advantages. Elimination of the secondary cooling system and water treatment system and the use of low carbon steel.

Primary Disadvantages. Lack of a heat sink in event of loss of coolant accident.

Possible future development work for water basin storage concepts include improvements in the economy of storage, evaluation of the consequences of extended loss of H_2O coolant, and possible design measures to provide secondary cooling by natural circulation of air.

17.2.2 Storage in Air-Cooled Vaults

17.2.2.1 Summary

Storage of bare (unpackaged) spent fuel in near-surface facilities with forced-draft cooling is an alternative for LWR fuel that has been out of the reactor at least 3 to 4 years. As shown in the process summary sheet, the facility would consist of reinforced Class I concrete storage vaults and auxiliary equipment to provide air circulation and to filter the effluent air. The spent fuel is stored vertically inside sleeves supported by top and bottom tube sheets that form plenums to distribute the inlet air and collect the effluent air.

Forced draft cooling is being used for HTGR fuels stored in closed but unsealed packages at INEL.⁽⁹⁾ Additional research and/or development efforts may be required to define radioactive confinement capabilities and requirements and to compare air-cooled storage to water-cooled basin storage from the standpoints of economics, safety, and maintainability.

17.2.2.2 Facility Description

Proposed air-cooled storage facilities for spent fuel assemblies are modular, each unit having the capacity for about 500 MT of spent fuel.⁽²⁾ In the forced-draft, air-cooled storage concept, the storage vaults are Class I reinforced concrete structures, partially buried in the earth. The structure provides biological shielding, protection from natural phenomena, and secondary confinement of radioactivity.

Concepts for air-cooled storage are shown in Figure 3 for fuel from the Canada Deuterium Uranium (CANDU) Reactor, and in Figure 4 for solidified high-level waste (HLW). Although these facilities illustrated are designed for natural-draft cooling, they resemble forced-draft cooled facilities in their architectural design, storage arrangement, and flow path of cooling air.

A high-efficiency filtration system is needed for the effluent air to provide a partial barrier to activity spread from failed fuel. The large resistance to air flow by this filter system would require a forced-air cooling system for the fuel. Natural convection cooling would not generate sufficient force for adequate cooling and air filtration; therefore, backup fans (excess capacity) would be designed with both normal and emergency power.

FIGURE 3. Concept for Air-Cooled Vault for Spent Fuel (Specifically for CANDU Fuel)⁽¹²⁾

FIGURE 4. RSSF Air Cooled Vault Concept⁽¹³⁾

17.2.2.3 Heat Removal

Cooling air is distributed in a bottom plenum, passes upward through the assemblies held vertically in sleeves, and exits at the top of the storage vault.⁽¹⁰⁻¹³⁾ A close fit of the assemblies within the sleeves ensures adequate distribution of the cooling air within the fuel assembly channels. Heat is transferred by convection to the air; some radiation to the sleeve effectively increases the heat transfer area.

Each 1500 MT of fuel stored in an air-cooled facility generates several megawatts of heat even after several years of storage. However, these heat loads are manageable with forced-air circulation. For example, after 5 years of cooling, an air flow of less than $25 \text{ m}^3/\text{sec}$ would adequately cool 1500 MT of fuel with only a 100°C rise in the air temperature. The surface area of the fuel rods is large, approximately $57 \text{ m}^2/\text{MT}$ for the composite assembly (Section 2.0). The surface heat flux after 5 years of cooling would be only 27 W/m^2 , resulting in low fuel sheath temperatures compared to temperatures of approximately 650°C in reactors.

If the primary cooling system fails, the heat capacity of the fuel and surrounding sleeves allows time to activate backup forced draft systems. Additionally, temporary bypass of the filter system would allow natural draft cooling of the spent fuel if the primary (forced draft) system fails but would result in higher than normal temperatures and increased risk of activity release.

17.2.2.4 Criticality Control

The spent fuel is maintained in a critically-safe array by the fuel storage sleeves. BWR assemblies arrayed on about 0.3-m centers and PWR assemblies on 0.5-m centers are critically safe even if flooded with water. These spacings can be reduced by inserting nuclear poisons in the storage array (e.g., use of borated steel for the sleeves). The facility design would meet applicable criteria for extreme natural phenomena.⁽²⁾ Strict procedural control and the use of instruments to measure fissile content would reduce the probability of an accident during handling prior to storage.

17.2.2.5 Corrosion

Air cooling of unpackaged spent fuel exposes stainless steel- and Zircaloy-clad fuel elements to air temperatures in excess of 100°C but probably less than 300°C . Storage of austenitic stainless steel-clad fuel elements in air at these temperatures presents no unusual problems if contaminants such as halogens and heavy metal ions are absent. Architectural use of this steel for 30 years⁽¹⁴⁾ has resulted in no unusual corrosion. Type 304 stainless steel has been used at temperatures of 816°C with an oxidation rate of only $2 \times 10^{-3} \text{ cm/year}$.⁽¹⁵⁾

At room temperature, massive Zircaloy is extremely stable to reaction with the common gases, but at a few hundred degrees centigrade, it reacts readily with oxygen, nitrogen, and hydrogen.^(16,17) Noticeable Zircaloy oxidation has been observed at 50°C, but oxidation ceases when a layer of the reaction product reaches a certain limiting thickness. However, stability of the surface oxide decreases with rise in temperature, and the film growth becomes rapid above 200°C.⁽¹⁸⁾

Extrapolation of rates of oxidation of Zircaloy between 350 and 500°C yields a rate of 2.5×10^{-5} cm/year at 100°C. A rate of 2.5×10^{-4} cm/year was obtained on zirconium heated in air at 200°C for 2800 hr.⁽¹⁹⁾ Storage of Zircaloy-clad fuel elements in air for extended periods may need further investigation to define the required temperature control. For extremely long-term storage, unknown factors include film stability and the amount of hydrogen pickup.

17.2.2.6 Radioactivity Monitoring and Confinement

In the event of a failure of the cladding on a fuel element stored without packaging, radioactivity would be released directly into parts of the storage facility. However, the cladding is designed to withstand a far more severe environment in the reactor than that normally encountered in a storage facility so that the probability of breach in the cladding is low.*

If the cladding were to fail and subsequently release fission products to the cooling air, prefilters and HEPA and carbon filters would remove about 99.9% of any particulates and about 99% of any iodine before the air is released to the atmosphere.⁽²⁰⁾ Noble gases and tritium may be released⁽²¹⁾ because treatment of such a large volume of air would probably be prohibitive. However, atmospheric dispersion of these radionuclides during normal operation is expected to reduce the offsite concentrations to levels below Federal guidelines. The facility would have radiation monitoring devices to detect and locate radioactivity releases. After leaking assemblies are located, they would be removed and packaged to confine the fission products.

A mechanism has been observed that would release radioactivity from leaking assemblies to the cooling air which is subsequently filtered to remove particulates. This involves the reaction of UO_2 with air to form U_3O_8 . The subsequent volume change could split the fuel cladding, further exposing UO_2 to the air.⁽²²⁾ This process is similar to the deliberate voloxidation of sheared fuel; at 450°C, 99% of tritium and 30 to 90% of noble gases have been released from sheared fuel test pieces within a few hours.⁽²¹⁾ A system to locate and remove leaking assemblies has not been designed but would minimize such a release.

* As low as 1% of the fuel fails during operation and another 1% is estimated to be subject to failure during subsequent storage.⁽⁵⁾

17.2.2.7 Maintenance

Continuous surveillance and continuously available maintenance would be required for equipment operation, activity monitoring, and removing and packaging failed fuel assemblies. Maintenance would be required to repair or replace inoperable components in the forced-draft cooling system, the radioactivity-monitoring system, the spent-fuel handling (transport, etc.) system, and the radioactivity-confinement system.

17.2.2.8 Status of Technology

17.2.2.8.1 Current. The basic technology to design an air-cooled facility that meets heat removal and criticality requirements already exists, and assemblies from gas-cooled reactors are being stored in air.⁽⁸⁾

17.2.2.8.2 Estimated Time of Readiness for Practical Design. Design of air-cooled vault storage facilities could proceed immediately. The required development (see below) could proceed during concept selection and initial design. The minimum time required to develop and construct an air-cooled storage facility was projected at 7 to 9 years in a preliminary study⁽¹⁰⁾ of air-cooled vaults for solid waste storage.

17.2.2.8.3 Major Development. Development may be required to:

- Characterize the fuel element temperatures with normal air flow and under accident conditions with natural convection cooling.
- Investigate the reaction of failed fuel in air storage at less than 300°C.
- Verify the corrosion resistance of Zircaloy for storage at expected operating conditions.

Process Summary

SPENT FUEL PACKAGING FACILITY

Facility Description. Class 1 structure for the receiving, packaging, and transferring spent fuel to storage.

Development Status. No technological barriers to the development of a packaging facility.

Confinement Barriers. Cladding and carbon and HEPA filters. [If water basins are used in the facility, filters and ion exchange columns (etc.) would be used to treat the water.]

Maintenance Required. General maintenance required for production type facility.

Waste Sources. Radioactivity that collects on filter systems for the cooling water and the ventilating air, and on other solid waste resulting from facility operation and maintenance.

17.3 PACKAGING OF SPENT FUEL

17.3.1 Summary

Packaging of spent fuel consists of sealing the fuel assembly (see Section 2.3.2 for fuel assembly description) inside a container after filling the container with a solid or gaseous heat transfer medium. The package is a second barrier for the confinement of fission products.

Spent fuel will probably not be packaged for storage until at least 3 to 4 years after the assemblies are discharged to allow decreased heat generation and radioactivity. The fuel is expected to be packaged at central storage facilities rather than at reactor sites because fuel reprocessing is most economical when central processing stations are used.⁽³⁾ Also, shipping cask volumes, and therefore shipping cask weights, can be minimized by shipping unpackaged rather than packaged assemblies.

Although packaged fuel assemblies from various reactors have been stored for about 10 years at the Savannah River Plant⁽²³⁾ and package concepts are currently being studied for CANDU fuel, additional development is required on packaging concepts for long-term storage of spent fuel.

17.3.2 Packaging Facility

A packaging facility consists of modules for:

- Receiving and handling spent fuel prior to packaging
- Receiving containers for spent fuel
- Packaging spent fuel
- Overpackaging failed fuel assemblies or spent fuel packages that have failed in storage
- Transferring packaged fuel to storage

The packaging facility and the facility for subsequent storage are expected to be at the same site. Fuel would be shipped from reactor storage basins in shipping casks designed for that purpose. The packaging facility would be expected to provide the same degree of protection for the spent fuel as is provided by the storage facility. Cooling is provided for the spent fuel throughout the facility.

A facility for packaging 1500 MT of spent fuel per year would require about 7000 single-assembly containers per year for BWR fuel or about 3000 single-assembly containers per year for PWR fuel.

17.3.2.1 Spent-Fuel Receiving Module

A variety of shipping casks containing spent fuel are delivered by truck or rail to this module. The casks are subsequently cleaned, decontaminated, and emptied. The empty casks are checked for abnormal conditions and then prepared for return. The fuel assemblies removed from the casks are cleaned,

monitored for failures, and transferred either to the Packaging Module or the Overpack Module, whichever is appropriate.

17.3.2.2 Container Receiving Module

The integrity of empty fuel containers is verified in this module.

17.3.2.3 Packaging Module

The fuel containers are thoroughly dried and loaded with spent fuel and filler material. The containers are then sealed and inspected before being sent to package storage.

17.3.2.4 Overpack Module

Any failed fuel assemblies discovered in the Spent Fuel Receiving Module are packaged as described in Packaging Module and then placed in a second container or "overpack" to provide two radioactivity confinement barriers. Any fuel containers found to be leaking are repaired and/or placed in "overpacks" to maintain two confinement barriers in the package.

17.3.2.5 Transfer Module

Containers are stored temporarily in this module before being transferred to the interim storage facility.

17.3.3 Fuel Assembly Container

17.3.3.1 Description

The simplest container is for single, intact BWR or PWR spent fuel assemblies. Although the use of containers that hold more than one fuel assembly is an economic alternative to using single-assembly containers, longer cooling times may be required before fuel packaging because of greater heat loads involved with more than one assembly. In addition, neutron absorbers may be required in multiple-assembly packages to ensure nuclear safety.

Physical size may restrict the number of assemblies that can be loaded into a single container. A square container with an inside dimension of about 44 cm could hold four intact PWR assemblies or nine intact BWR assemblies. This package would weigh about 3 MT.

An alternative to packaging of the intact spent fuel assembly is dismantling the fuel assembly and storing the fuel elements separate from the non-nuclear materials. The volume of the fuel elements is about 40% of that occupied by the intact assembly; thus the number and/or size of spent fuel containers is potentially smaller. If criticality (rather than heat transfer) limits package loading, neutron poisons could be included in the package.

17.3.3.2 Materials

The container material depends on the repository in which the package is to be stored and cooled. Highly corrosion-resistant material⁽²⁴⁻³⁰⁾ such as

Type 304L stainless steel, Type 310 stainless steel, Zircaloy 800, or "Inconel"[®] 625 have been considered for containers stored in water basins.⁽¹⁰⁾

Low carbon steel is an economic alternative material to stainless steels, Zircaloy, or "Inconel" when fuel storage is in air or when moisture can be excluded from the container.⁽²²⁾ Low carbon steel corrodes about 0.005 cm/year at 260°C when shielded from the weather (air-cooled vault). A wall thickness of 1.3 cm provides an adequate factor of safety against corrosion for 100-year storage.

Aluminum is particularly attractive as a container material for storage in air-cooled vaults because of the protective oxide film formed on aluminum in air. The workability of aluminum is also advantageous.

The container will be subject to radiation during storage. However, embrittlement of either steel or aluminum is not expected to occur at a neutron dose lower than 10^{21} n/cm². For 100-year storage, the neutron flux required for embrittlement would be $>10^{11}$ n/(cm²-sec). This flux is several orders of magnitude greater than the neutron flux from spontaneous fission in discharged spent fuel.

17.3.4 Heat Removal

When fuel assemblies are enclosed in a container, the heat generated must be transferred from the fuel to the container walls and ultimately to the environment. The final temperature of the packaged fuel assembly will be the container temperature plus the temperature rise from the container wall to the hottest part of the fuel. The additional step of transferring heat from the fuel to the container will cause the fuel assembly to be hotter than during unpackaged storage. This internal temperature rise can be reduced by using a material with high thermal conductivity to fill the space between the fuel rods.

Helium, which is most frequently considered as the filler material for packaged fuel assemblies^(12,22,31) has a thermal conductivity of 0.17 W/(m-°C) at 100°C [the thermal conductivity of air at the same temperature is 0.035 W/(m-°C)]. In a helium-filled package, part of the heat is transferred from the fuel elements to the container by thermal radiation; the amount depends on the fuel element temperature. In addition to its simple and economical use, helium provides an inert atmosphere for storage of the spent fuel, thereby reducing the possibility of corrosion-induced failure of fuel cladding.

Zinc (melted, poured, and solidified) has been studied as a filler for some CANDU fuels.⁽¹²⁾ With a thermal conductivity of about 110 W/(m-°C) a zinc filler would maintain substantially lower fuel element temperatures than would a helium atmosphere.

[®] Trademark of International Nickel.

Powdered aluminum in a helium atmosphere is a technically simpler filler to use than molten and solidified zinc. The thermal conductivity of aluminum [about 200 W/(m-°C) for solid metal] is reduced somewhat in the powder form, but the effective thermal conductivity of aluminum powder with 60% packing density and filled with helium equals or exceeds the conductivity of solid zinc.

The only other material normally mentioned for use as a filler material is air. Liquids in general are excluded as filler material because of the possibility of vaporization or decomposition to gases with consequent rupture of the container.

The upper limit suggested for the temperature of stored solidified HLW to avoid changes in form or evolution of decomposition products is the maximum waste temperature reached at the reprocessing plant.^(10,32) If the same reasoning is applied to storage of spent fuel, the temperature limit in the absence of air is the maximum obtained during operation in the reactor, or about 650°C.

17.3.5 Packaging Failed Fuel

Some fraction of the spent fuel assemblies presented for packaging will contain fuel elements with breached claddings. A second container or "overpack" placed over the initial container would restore core confinement to approximately the degree that was provided by the container and the unbreached cladding. The use of an overpack is planned for solidified HLW containers that fail.⁽¹⁰⁾

A solidified metal filler in the fuel container would, in addition to improving heat transfer characteristics, provide secondary confinement for failed fuel assemblies. The cost of using a filler solely for secondary confinement (i.e., if it is not required for heat removal) should be compared to the cost of a second container.

During storage of the packaged fuel assemblies, there is a small but finite probability that leaks will develop in both the container and fuel cladding with subsequent escape of fission products from the package. Repair of the leaking container followed by enclosure in a second container (overpack) will restore double confinement. If repair of the primary container is not possible, the fuel assembly is removed, repackaged, and "overpacked."

17.3.6 Radioactivity Monitoring

The possibility of leaks from the containers, necessitating container repair or fuel repackaging, suggests that the storage facility include monitoring systems that can first detect leaks and then locate the leaking container.

Monitoring within individual containers to detect breach of the fuel cladding and fission product release is impractical because the container is a confinement barrier. A monitor system which penetrates the container increases the likelihood of container failure. Monitoring and control of radioactivity would be included in the storage facility operation. If helium is used as a filler material in fuel containers, helium detectors may detect leaks before fission products are released.

17.3.7 Status of Technology

17.3.7.1 Current

Most of the available technology concerning packaging of spent fuels has been developed for fuels other than those from LWR's. Conceptual designs have been made of facilities for packaging of CANDU spent fuels,⁽²²⁾ FFTF spent fuels,⁽³¹⁾ and canisters of solidified HLW.⁽¹⁰⁾ A facility for packaging and storing fuel from the Fort St. Vrain and Peach Bottom high-temperature gas-cooled reactors and from the Rover nuclear rocket program has been constructed at the Idaho Chemical Processing Plant.⁽⁹⁾

AECL has identified several packaging concepts for use with CANDU fuels.^(12,22) Three of the concepts involve encapsulation of fuel in mild steel containers under helium atmospheres (Figures 5, 6, and 7). All three package types are intended for use with surface silos but could be used with other storage concepts. In the fourth concept (Figure 8), six fuel assemblies are placed in a trefoil-shaped aluminum extrusion with zinc or aluminum as a filler material.⁽¹²⁾ In all of the above concepts, a helium filler protects against corrosion of fuel cladding or the UO_2 -air reaction in failed elements.

A concept developed for the FFTF fuel⁽³¹⁾ calls for a helium-filled stainless steel container for each fuel assembly.

Graphite fuel assemblies from the Peach Bottom or Fort St. Vrain reactor are packaged in unsealed containers of 0.635-cm-thick carbon steel. The containers are 45.7 cm in diameter and 335 cm long. They are stored in a forced-draft-cooled facility at the Idaho Chemical Processing Plant.⁽⁹⁾

Packaged fuel assemblies from several different reactors, including Hallam and Dresden, have been stored underwater at the Savannah River Plant for several years.^(23,31)

17.3.7.2 Estimated Time of Readiness for Practical Design

There are no technological barriers to the design of spent-fuel packages and packaging facilities. However, it is estimated that two to four years would be required to verify the corrosion resistance of container materials, perform the required calculations, and optimize container design and loading for the storage concept. The design and construction of a packaging facility

could proceed concurrently with that of a storage facility with an estimated elapsed time of seven to nine years.

17.3.7.3 Major Development

Identified development areas include:

- Selection of construction material for fuel element containers. Materials selection should consider a cost benefit analysis for the container in a particular storage medium or concept.
- Selection of container filler. If a material other than helium is selected, the compatibility of the filler with metals over long periods of time should be investigated.
- Generic calculations for both PWR and BWR fuel elements (intact and disassembled) to determine allowable package loading considering both criticality and heat removal.

FIGURE 5. Schematic Showing Packaging Concept for CANDU Fuel⁽¹²⁾

FIGURE 6. Schematic Showing Packaging Concept for WR-1 (AECL) Fuel⁽²²⁾

FIGURE 7. Square Basket for Douglas Point (AECL) Fuel⁽²²⁾

FIGURE 8. Packaging Concept for CANDU Fuel⁽¹²⁾

Process Summary

WATER BASIN STORAGE — PACKAGED SPENT FUEL

Storage Description. Near-surface, water-filled cell of heavily reinforced concrete lined with stainless steel.

Development Status. Demonstrated by about 30 years experience (unpackaged fuel) at reactor sites and fuel reprocessing plants.

Heat Removal. Basin water circulation with rejection to atmosphere.

Materials Used. Concrete and stainless steel.

Confinement Barriers. Cladding, package, basin water, and carbon and HEPA filters.

Maintenance Required. Filtration systems, heat removal systems, fuel assembly transfer systems, and radiation monitoring systems.

Waste Sources. None normally, if package and cladding fail radioactivity collected by the filter systems and on other solid waste resulting from facility operation and maintenance.

Primary Advantages. Low fuel temperatures, the additional confinement barriers, the large heat sink and transparent radiation shield provided by the water.

Primary Disadvantages. Requirement for stainless steel basin liner and spent fuel container, for water quality control, and a substantial water supply.

17.4 STORAGE OF PACKAGED SPENT FUEL

17.4.1 Storage in Water-Cooled Basins⁽²⁾

17.4.1.1 Summary

Water basin storage of packaged LWR fuel assemblies, as shown in the process summary sheet, provides container cooling and radiation shielding while allowing visual inspection and direct manipulation of the package. Modular basin construction allows facility expansion with a minimum of additional support facilities and services. The storage basin can also serve as a large heat sink under some accident conditions, and the water normally retains any escaping particulate radioactive material until it can be removed by filters.

17.4.1.2 Facility Description

The spent fuel storage basins would be stainless steel-lined concrete and would presumably be located adjacent to the fuel packaging facility from which packaged fuel assemblies could be transferred, either in water-filled casks or via a transfer canal. Design requirements are essentially the same as previously described for water basin storage of bare fuel assemblies (Section 17.2.1). The packages would be placed in storage racks within the basins similar to those in which fuel assemblies are stored in fuel storage basins at reactor sites.

In the concept for water basin storage of canisters of solidified HLW,⁽¹⁰⁾ 500 waste canisters would be stored in double-tiered racks in each module. An equivalent (equal heat load) design for packages of spent fuel would store 3000 single-assembly PWR packages or 7000 single-assembly BWR packages in each module. Because of nuclear safety requirements, each spent-fuel storage module could require up to about 400 m² (double-tiered storage of unpoisoned single-assembly PWR packages on 0.5-m centers) compared to about 60 m² suggested for storage of 500 waste canisters.⁽¹⁰⁾

The basins contain demineralized water to cool the packages, control container corrosion, provide visibility for inspection, and provide shielding and containment. The large number of fuel packages may require simultaneous charging of several storage basins. The basin building houses the handling equipment and protects the facility from abnormal weather.

17.4.1.3 Heat Removal

Decay heat from the spent fuel is removed by the continuous circulation of the storage basin water over the fuel assembly packages; heat is then rejected to the environment through a heat exchanger and secondary coolant. The system maintains the basin water temperature low enough to control container corrosion and to prevent algae growth. The facility design would normally allow rapid replacement of this equipment during an emergency.

Loss of cooling to a freshly charged cell could result in water boiling within 3 to 4 days. In about 6 weeks a cell would boil dry. Water replacement and early repair of water cooling equipment would correct the situation.

17.4.1.4 Water Treatment

Each basin should have its own cooling system and its own water purification system; all basins in a storage facility should be isolated from each other. A small filter-demineralizer system can then maintain water purity and clarity and remove any radioactive material that may escape from a container and enter the basin water. In the event of gross water contamination caused by a leaking package (10-year-old PWR assemblies contain over 100,000 Ci of fission product activity), a portable filter-demineralizer unit of much higher capacity than normally needed could be put into service to decontaminate the water.

If stored fuel containers are fabricated from 304L stainless steel, the basin-water treatment system would be designed to maintain chloride ion concentrations below 5 ppm to avoid stress-corrosion cracking. Cleaning fuel assembly containers and any process equipment added to the basins prior to insertion would help to maintain the basins almost chloride free. Proper control of water chemistry may ensure package integrity for at least 100 years. Literature data indicate negligible corrosion of general 304L stainless steel at temperatures near 100°C (corrosion rates range from 0.0002 to 0.002 cm/year). (30,33-37)

17.4.1.5 Criticality Control

The storage racks that hold fuel packages are expected to be patterned after the racks designed for storing fuel assemblies in reactor storage basins with the assurance of criticality control built in and adequately demonstrated. BWR and PWR assemblies require storage on not less than 0.3-m and 0.5-m centers, respectively, to maintain a critically safe array using unpoisoned storage racks.

17.4.1.6 Radioactive Monitoring and Confinement

The water in the storage basin constitutes an additional confinement barrier should radioactivity escape through leaks in both the fuel cladding and the package. Use of a secondary cooling loop to transfer the heat from the basin water to the environment minimizes the release of radioactive materials if packages leak and contaminate the basin water. Radiation detection equipment installed in the cooling water recirculation line for each cell would give early warning of a leaking package. Portable radiation detection equipment could be used to locate the leaking package so that the package could be quickly removed for repair and "overpacking." Ventilation air may normally be discharged directly to the atmosphere through a stack;

Process Summary

AIR-COOLED STORAGE — PACKAGED SPENT FUEL

Storage Description. Near-surface, heavily-reinforced concrete structure (Class 1).

Development Status. The basic technology demonstrated by an existing facility for air-cooled storage of packaged (unsealed) assemblies from gas-cooled reactors.

Heat Removal. Natural circulation of air.

Materials Used. Concrete and low carbon steel.

Confinement Barriers. Cladding and package. [Provisions could be made for forced draft cooling in the event of radioactivity release and additional confinement protection provided by filters (HEPA and carbon) in the forced air cooling system.]

Maintenance Required. Radiation monitoring system and fuel transport equipment. (If emergency forced draft cooling is provided, maintenance of this system would also be required.)

Waste Sources. None normally, if package and cladding fail, radioactive materials could enter the cooling air stream. (If emergency forced draft cooling is provided in the event of an activity release, the material other than tritium and the noble gases would be collected in the filter system.)

Primary Advantages. Elimination of a secondary cooling system and water treatment system, the use of a passive coolant system and low carbon steel.

Primary Disadvantages. Lack of a heat sink in event of loss of coolant accident.

however, in the event of activity escape, as indicated by air monitoring instrumentation, exhaust air could be automatically diverted through HEPA and carbon filters before discharge.

17.4.1.7 Maintenance

The water treatment and air ventilation systems will require routine maintenance to ensure proper operation of pumps and fans. Periodic replacement of filters and ion exchange resin would be required. This storage concept would require surveillance while fuel assemblies are in storage.

17.4.1.8 Status of Technology

17.4.1.8.1 Current. Pool storage of packaged fuel assemblies is an adaptation of the current spent fuel storage technique used by all LWR operators. Water for storing and shielding fuel elements has been used since the first Hanford reactors began operation in the early 1940's. Packaging of fuel assemblies prior to water storage has been practiced at the Receiving Basin for Offsite Fuels (RBOF) at the Savannah River Plant⁽²³⁾ for about ten years.

17.4.1.8.2 Estimated Time of Readiness for Practical Design. Design of a central water storage basin for spent fuel assemblies could begin immediately. The storage facility would likely be modular in design with additions being added as necessary to meet increasing inventory requirements. The first modules could be ready in 6 to 8 years.

17.4.1.8.3 Major Development. No major development work is expected to be required prior to implementation of basin storage of spent fuel assemblies in packages. The most probable mode of package failure may be by container corrosion. Studies of corrosion of stainless steels in water would continue. Assessment of alternative packaging materials and improvements in the economy of storage would also continue. An assessment of the consequences of extended loss of coolant water may be required, including possible design measures to provide secondary cooling by natural circulation of air.

17.4.2 Storage in Air-Cooled Vaults^(10,13)

17.4.2.1 Summary

After irradiated fuel has adequately cooled in a water basin, the fuel assemblies could be packaged in metal containers, which could then be stored in a concrete vault. In a properly designed facility, the decay heat from residual fission products would create a natural draft sufficient for cooling the package. The cooling is passive and should require little maintenance and only minimal surveillance. If facility design is based on the modular concept, additional storage space could be added as fuel inventory increases. An air monitoring system would detect leaking containers, which would then be

removed for repackaging. In current designs proposed for natural convection storage vaults, coolant air is not continuously filtered. Some release of fission products to the environment could occur if both the fuel assembly cladding and the container fail. The release could be minimized if a standby forced-air cooling system with HEPA and carbon filter were activated.

17.4.2.2 Facility Description

Reinforced Class I concrete structures partially buried in the ground have been conceived as air-cooled storage vaults (Figure 9). The vault is divided into cells, each cell is capable of storing several hundred spent fuel assemblies. Design requirements are essentially the same as those discussed for air-cooled vault storage of bare fuel assemblies (Section 17.2.2). According to a typical design, the packaged fuel assemblies are stored vertically and can either be suspended from the roof of the cell or supported by posts placed in the cell floor. Metal sleeves are provided for each storage position. The sleeves are attached to tube sheets at each end of the sleeves; the tube sheets form plenums for distributing the incoming air to the sleeves

FIGURE 9. RSSF Air-Cooled Vault Concept⁽¹⁰⁾

and directing the effluent air to the vault exit. Packaged fuel assemblies are charged into the storage position by removing a massive concrete plug from the roof above each sleeve. The plug is replaced after charging. A bottom-loading shielded transporter moves the container from the packaging facility to the storage vault. The large number of packages would probably require simultaneous charging of several cells in any one vault.

In the air-cooled vault concept, the decay heat of the spent fuel and the design of the vault combine to induce air flow by natural draft. The heated air is discharged through a short exhaust port to the atmosphere without being filtered. Monitoring systems are designed to detect air flow restrictions or interruptions and the presence of airborne radioactivity. Repair or overpacking leaking containers would reduce activity releases.

17.4.2.3 Heat Removal

In the proposed concepts for air-cooled vault storage, fission product decay heat is removed from the fuel containers by natural convection. Air entering the bottom of the vault is heated in the annulus between the storage sleeve and container, rises to the upper plenum, and exits through a short stack. The temperature of the coolant air is determined by vault design and by age of the stored fuel assemblies. Initial exhaust temperatures of about 100°C will probably be experienced. This temperature decreases with time as the fission products decay.

17.4.2.4 Corrosion

For air-cooled vault storage, an initial container surface temperature of more than 100°C has been calculated. This temperature precludes condensation of moisture on container surfaces and is expected to limit uniform corrosion of the container, perhaps to as low as 0.005 cm per year. The actual corrosion rate depends on the container material and atmospheric impurities. A stainless steel can would be expected to corrode at a lower rate than carbon steel; however, the stainless steel container may be too expensive to justify the extra cost. Treatment of the inlet air to reduce humidity and remove corrosive gases and halides might be required in some locations.

17.4.2.5 Criticality Control

The structural requirements of a vault providing individual storage compartments for each packaged fuel assembly have resulted in spacing between packages that is adequate to provide control of criticality in all studies made thus far. Fuel assemblies may be spaced in the vault at sufficiently greater distances than those between assemblies stored in water basins to ensure a critically safe arrangement even if the vault were flooded.

17.4.2.6 Radioactivity Monitoring and Confinement

An exhaust air monitoring system would detect releases of radioactivity from the stored fuel assemblies. Because coolant air moves through the vault by natural rather than forced draft, the buoyant force may be too low to pass sufficient quantities of air through high-efficiency filters. In the absence of filters for exhaust air, effective containment of fuel assembly core material depends on the integrity of cladding and container. A backup forced draft system equipped with exhaust air filters could be activated by radiation monitors and thereby effect partial activity confinement. Even if the exhaust air is filtered, a defective package would be removed from the vault, repaired and "overpacked," and replaced in storage in a special vault designed for doubly contained assemblies. A failure rate of one percent each for the fuel cladding and container would result in only one failure of both containment barriers for every 10,000 packages. Thus, the special vault for failed packages would be quite small.

17.4.2.7 Maintenance

Cooling by a natural air draft eliminates maintenance of large banks of blowers, fans, coolers, and heat exchangers. Monitoring instruments and equipment for charging and discharging packages to and from the vault would require periodic inspection, maintenance, and/or replacement. Inlet and effluent air ports would also require periodic inspection to ensure free flow of air. For the first several years of storage, a program of routine inspection could be adopted to assess the integrity of the container. The need for inspections would presumably decrease should experience show that container integrity was not diminishing with time.

17.4.2.8 Status of Technology

17.4.2.8.1 Current. Spent fuel elements from gas-cooled reactors are being stored in unsealed packages in forced-air-cooled vaults.⁽⁸⁾ LWR fuel assemblies are not currently stored in air. Remote handling of large quantities of radioactive materials in massive high-quality reinforced concrete structures used in reprocessing plants provides the basic technology available and required and may be adapted to storage vaults.

17.4.2.8.2 Estimated Time of Readiness for Practical Design. Air-cooled vault concepts have been designed by ARHCO^(10,11) for solidified waste. This or similar vault designs could be adapted to the storage of packaged fuel assemblies. Spent fuel would probably be stored in a reactor basin for up to ten years prior to storage in air-cooled vaults. If a decision to apply the air-cooled vault concept were made in the near future, design and construction of a facility would require an estimated seven to nine years.

Process Summary

CONCRETE SURFACE SILO — PACKAGED SPENT FUEL

Storage Description. The spent fuel is sealed in a steel container which is placed in a concrete housing and stored vertically outside on a concrete pad.

Development Status. Technology is being developed now for CANDU fuels.

Heat Removal. Natural circulation of air between the steel cask and the concrete housing or conduction through the cask and housing to the atmosphere.

Materials Used. Concrete and low carbon steel.

Confinement Barriers. Convection cooled concept; cladding and package. Conduction cooled concept; cladding, package, and steel liner.

Maintenance Required. Radiation monitoring and fuel transport systems.

Waste Sources. None normally, if cladding and package fail, the package, and concrete silo could become radioactive wastes.

Primary Advantages. Low maintenance requirements, massive confinement barriers in addition to the fuel cladding, and totally passive coolant system.

Primary Disadvantages. Large land use and material requirements.

17.4.2.8.3 Major Development. The concept of natural-draft cooling requires development of little new technology. Development studies may be required to better define details of confinement and design. Packaging development for this concept could concentrate on corrosion studies of candidate container materials in a wide range of possible environments (climatic conditions, pollutants) to match the package to the site of the storage vault.

17.4.3 Storage in Concrete Surface Silos^(10,13,22)

17.4.3.1 Summary

Concepts of storing spent fuel in surface-located concrete silos are being studied. In one concept,⁽²²⁾ the spent fuel assemblies are packaged in 5-cm-thick, low-carbon steel containers, placed in concrete silos which serve as neutron shields, and stored vertically on outdoor concrete pads. The annulus between each package and its concrete silo serves as a chimney through which cooling air passes by natural convection.

In a second concept, the spent fuel package is placed directly in contact with a steel and lead-lined concrete silo and stored vertically on outdoor concrete pads. A welded lid on the steel liner produces a third confinement barrier. Heat is dissipated by conduction through the lead and concrete. This concept has been proposed for relatively low-burnup (<10,000 MWD/MT)⁽²²⁾ CANDU-type fuel and might not be practical for storing the higher burnup (average 25,000 MWD/MTHM, maximum ~33,000 MWD/MTHM) LWR-type fuels.

17.4.3.2 Facility Description

Although fairly firm designs have been made for storing heavy-water power reactor fuel in concrete surface silos,⁽²²⁾ no such designs have been developed for LWR fuel. Concrete surface silos for storing solidified high-level wastes have been designed,⁽¹⁰⁾ and these designs could be modified to adapt this concept to storage of spent LWR fuel elements.

The large number of fuel assemblies to be stored (about ten times as many as canisters of solidified waste) suggests that multiple packaging would be desirable to minimize the number of concrete silos required. The fuel assemblies would be welded inside 5-cm-thick carbon steel container.* The packages would be stored either in convective-cooled or sealed conductive-cooled concrete housings which provide radiation shielding. The natural draft-cooled concept (Figure 10) could store up to 6 PWR or 15 BWR fuel assemblies in each cask, which would generate about 3 kW of decay heat after 10 years cooling. The conductive-cooled concept (Figure 11) would probably contain fewer assemblies per storage unit. One-meter-thick concrete walls would reduce radiation dose rates outside the storage unit to about 2 mrem/hr.

* An alternative to packaging the spent fuel in the 5-cm-thick cask is to first package the fuel in a thin walled container followed by placement in the cask.

FIGURE 10. Concrete Surface Silo, Convection Cooled Concept⁽¹⁰⁾

FIGURE 11. Concrete Surface Silo, Conduction Cooled for CANDU Fuel⁽⁴⁴⁾

17.4.3.3 Heat Removal

Fission product decay heat and the design of the air-cooled storage unit induce air flow by natural draft to maintain acceptable package and fuel assembly temperatures. Cool outside air enters through openings in the base of the silo, passes up through the annulus between the package and silo, and leaves through openings near the top of the silo. With a decay heat load of about 3 kW and an inlet air temperature of 25°C, the exit air temperature would be about 40°C. The cask surface temperature would be almost 100°C.

The conductive-cooled storage unit would operate with a surface temperature between 50 and 100°C depending on the time of day, ambient air temperature, and extent of exposure to sunlight. The inner-cask wall temperature would be 100 to 200°C, and the fuel assembly temperature would be 300 to 400°C. All temperatures are approximate and would, of course, depend on the final design.

17.4.3.4 Corrosion

The corrosion rate of low carbon steels in air is about 0.005 cm/year at 260°C. With a container wall thickness of 5 cm, convection-cooled concrete surface silo storage for several hundred years would be safe. Conduction-cooled packages should not corrode because there is no exposure to air. The concrete silo would gradually erode but would be expected to last for several hundred years.

17.4.3.5 Criticality Control

Single fuel assemblies in concrete surface silos would present no criticality potential; however, practical considerations would require multiple packaging. The thick concrete shields would protect against occurrences of criticality from too close spacing of storage units.

17.4.3.6 Radioactive Monitoring and Confinement

Fission products and actinides contained in spent fuel cores are protected from accidental release by multiple barriers. In the conduction-cooled concept the fuel cladding, the package, and the sealed steel liner aid in preventing the contents of the spent fuel from reaching the environment. In the convection-cooled concept, two barriers, the cladding and the package, provide confinement of the radioactivity. The storage unit is designed to maintain confinement under all credible conditions and is capable of withstanding forces from natural phenomena and/or accident conditions.⁽²⁾ Loss of confinement would be detected by area survey instruments and by observations of the storage array.

Process Summary

Storage Description. Storage is in lined holes in mined salt (or other geologic formation) deposits.

Development Status. Technology being developed for storage of solidified HLW. Additional development is required on stability of geologic formation (for retrievability).

Heat Removal. Conduction through the surrounding formations, eventually to atmosphere.

Materials Used. Concrete and low-carbon steel.

Confinement Barriers. Cladding, package, and carbon and HEPA filters.

Maintenance Required. Ventilating draft system including power supply and filters (HEPA and carbon), radiation monitoring system, and the fuel transport systems.

Waste Sources. None normally, if cladding and package fail radioactivity collected by the filter system and on solid wastes resulting from facility operation and maintenance.

Primary Advantages. Totally passive cooling, the extensive radiation barrier, and the third confinement barrier (in addition to fuel clad and container).

Primary Disadvantages. Difficult construction and possible geologic instabilities.

17.4.3.7 Maintenance

Routine maintenance of storage described by this concept is not normally required. Periodic inspection would be required to ensure free flow of coolant for the convection-cooled storage units. Monitoring instruments would require periodic inspection and replacement to warn of incipient release of radioactive material. Service facilities would receive routine maintenance.

17.4.3.8 Status of Technology

17.4.3.8.1 Current. A project to demonstrate conductive-cooling storage of spent fuel is currently being planned in Canada.⁽²²⁾ Packaged spent fuel from D₂O-moderated reactors will be placed in lead- and steel-lined hollow reinforced concrete silos, which will be stored outdoors on concrete pads at the Whiteshell Nuclear Research Establishment. Structural integrity of the storage unit will be monitored closely during the demonstration period. No program is currently planned to demonstrate concrete surface silo storage of LWR fuel using either conductive or convective cooling.

17.4.3.8.2 Estimated Time of Readiness for Practical Design. Construction schedules developed by ARHCO⁽¹⁰⁾ indicate that concrete surface silo storage facilities for storage of waste from a reprocessing plant could be ready for operation within about 7 years. Storage of spent fuel according to the same concept might be available on a similar schedule.

17.4.3.8.3 Major Development. Development would likely be required to ensure that welding and nondestructive-testing operations conducted under semiremote conditions can meet necessary quality assurance standards and assure the long-term integrity of the package.

17.4.4 Storage in Geologic Formations^(21,32,38-40)

Storage of spent fuel in geological formation (e.g., salt mines) will be similar to waste storage that is treated in detail in Section 26.2 and therefore is not discussed here. Interim storage in geologic formations could eventually become a mode of disposal for both spent fuel and high-level waste. Proper initial design would facilitate this change.

The area required for storage of fuel will be about the same as that for waste if heat dissipation determines spacing (such as in salt beds). Fuel will require more area if the number of packages determines the storage area.

Unlike waste-storage mines, mines for fuel storage have the potential for releasing fission gases and tritium. Removing these gases from the circulating air and/or sealing the filled mine sections to minimize the amount of air to be treated are alternatives to reduce releases.

Process Summary

NEAR SURFACE HEAT SINK STORAGE — PACKAGED SPENT FUEL

Storage Description. The packaged spent fuel is contained in lined holes in the earth's surface, covered with a concrete plug.

Development Status. Concept currently being tested for some CANDU fuels.⁽⁴³⁾

Heat Removal. Conduction and convection through the surrounding earth to the atmosphere and the water table.

Materials Used. Concrete and low-carbon steel.

Confinement Barriers. Cladding and package (some additional confinement of particulates is provided by the hole liner and plug).

Maintenance Required. Radiation monitoring system, and the fuel transport system.

Waste Sources. None normally, if cladding and package fail contaminated hole liners and/or packages could be a source of radioactive waste.

Primary Advantages. Low maintenance requirements, ease of construction, and totally passive cooling system.

Primary Disadvantages. Large land use and material requirements.

17.4.5 Near-Surface Heat Sink Storage

17.4.5.1 Summary

Placing spent fuel packages into lined holes in the earth's surface provides storage with passive cooling and stable shielding. This low maintenance concept is being tested for CANDU fuel.⁽⁴¹⁾ Storage sites of this type could probably be constructed and begin receiving fuel sooner than would facilities based on any other concept.

17.4.5.2 Facility Description

In this concept, packages containing one or more fuel elements are placed in lined holes, and covered with a plug to provide radiation shielding and permit access to the area. The concept is similar to the method used for storage of some CANDU fuels at Chalk River, Canada (Figures 12, 13, 14). The spacing of the holes is determined by criticality requirements and by the heat load that can be dissipated throughout the surrounding medium. The hole liner and the shield plug provide protection against entry of ground water or surface water. Materials (such as concrete) that are resistant to corrosion attack from the surrounding earth are used for the hole liner.

FIGURE 12. Typical Concrete Tile Hole for Surface Storage of CANDU Fuel⁽⁴¹⁾

FIGURE 13. Typical Array of Tile Holes, CANDU Fuel⁽⁴¹⁾

FIGURE 14. Tile Holes⁽⁴¹⁾

The water table at the storage site should be sufficiently lower than the bottom of the liner hole so that during any credible rise in the water table, ground water cannot reach the liner. This is easily accomplished in arid regions with unsaturated zones 100 to 600 m thick.⁽⁴²⁾

17.4.5.3 Heat Removal

The heat generated in the spent fuel elements is dissipated through the surrounding earth to the atmosphere and to the water table.^(42,43) The storage holes are spaced sufficiently far apart to allow heat removal without excessive fuel element temperatures or generation of steam with the resultant pressures.

17.4.5.4 Corrosion

A corrosion-resistant material for the hole liner and shielding plug, such as concrete, avoids corrosion problems that might occur with metal liners. The spent fuel package is sealed from the outside environment. Contact only with dry air and concrete reduces container corrosion.

17.4.5.5 Criticality Control

Criticality is controlled by properly spacing the storage holes prepared for the spent fuel packages. BWR and PWR fuel elements are critically safe when arrayed on 0.3-m and 0.5-m centers, respectively, in water. The spacing required for cooling spent fuel packages in earth will usually exceed this critically safe spacing.

17.4.5.6 Radioactivity Monitoring and Confinement

The confinement barriers for this method of storage are the fuel cladding, the fuel package, and the hole liner and shielding plug. Radioactivity monitors throughout the storage site could locate the leak site if a release occurs. Repair or overpackaging of leaking packages would reduce radioactivity releases.

17.4.5.7 Maintenance

Maintenance is required for repair of monitoring equipment and repair or overpackaging of failed containers. Replacement of the concrete hole liners may be required after very long storage periods because of the deterioration of concrete.

17.4.5.8 Status of Technology

17.4.5.8.1 Current. Most of the technology for development of near-surface storage of packaged spent fuel elements is available. The principles are known, and most of the design requirements are calculable or can be mocked up at proposed sites. Development to date is limited to Canadian tests with CANDU fuels⁽⁴¹⁾ and the similar concepts for storage of HLW.⁽⁴²⁾

17.4.5.8.2 Estimated Time of Readiness for Practical Design. Design of a practical near-surface storage facility is currently feasible. However, design optimization and assurance of design lifetime may require concurrent material studies.

17.4.5.8.3 Major Development. Development includes the characterization of:

- Heat transfer from a widely distributed heat source (spent fuel elements) located in a near-surface heat sink.
- The effect of the heat source on the storage medium.
- Specific studies of long-term deterioration of storage liners in the selected site.

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18.0 INTERIM STORAGE OF CHOP-LEACH FUEL BUNDLE RESIDUES

18.0 INTERIM STORAGE OF CHOP-LEACH FUEL BUNDLE RESIDUES

18.1 CHARACTERISTICS OF PRIMARY WASTE CONTAINMENT AND INTERIM STORAGE

The main objective of interim storage of fuel cladding waste is to isolate the materials from man's environment until these wastes are transferred to a treatment facility or a permanent disposal site. The major technologies include those associated with the container, vault or burial structure; monitoring between barriers; retrieval from storage, transportation to the final storage site; and shipment to the final disposal site. This section discusses current handling practices of fuel cladding from LWR fuels and describes developmental work needed if future reprocessors are to use a shielded structure for storage. The properties of the waste stream are discussed in Sections 2 and 7. Both the chemical and radioactive properties of the waste need to be carefully considered in storage technology. Transportation of hulls between sites is discussed in Section 22.

18.1.1 Containers

Conventional, thin-gauge, carbon steel barrels of approximately 115-liter capacity are the containers which have been used to bury fuel residues.⁽¹⁾ The top is held in place by spring-loaded clamps. At the nominal throughput of 1 MTU/day, the plant loads about 3 barrels a day.

Plans have been made to use nominal 2300-liter containers (1.1 m diam by 2.4 m long) fabricated from 3-mm thick 304 stainless steel for interim storage. A bolted closure has been provided. At a nominal plant throughput of 5 MTU/day, the cladding fills one of these containers a day.

Containers planned for use at the BNFP plant have provisions for gasketed covers and vent holes to release pressure. This pressure could result from evaporation of residual moisture from the hulls rinse operation, the evolution of gases from the radiolysis of residual moisture, and corrosion-produced hydrogen.

Future reprocessing facilities should be able to devise additional container designs adapted for transport and storage operations. They will probably use stainless steel as the material of choice for the waste container. In support of future reprocessing plants the following development needs are presented.

1. Determine the time required to evaporate residual moisture from cladding waste by radioactive decay heat of activation and fission products.
2. Develop additional techniques to make final closure on hulls containers and methods of testing closure integrity.

3. Develop additional methods to remove radioactive contamination from the waste container, or
4. Develop a system to provide additional confinement over the contaminated waste container.

Placing of fuel cladding and associated scrap in underground vaults of suitable design was intended⁽²⁾ in the fuel reprocessing plant at Morris, Illinois. The design assumed that the cladding and scrap would be discharged into a water-filled, stainless steel-lined concrete vault with leak monitoring capability. If regulations requiring the transfer of alpha-contaminated wastes to an approved repository were established, it would be necessary to provide for eventual removal of the cladding from the water-filled vault for shipment to another repository. There would be no unique gains from such an operation, and the physical handling of the cladding waste would be significantly more complex than direct placement in a transfer container prior to interim storage.

18.1.2 Vault Storage

The containment for the cladding hulls containers in future plants could be a concrete vault. The size of the vault will reflect how long the reprocessor plans to provide storage prior to treatment and/or prior to permanent storage. Some of the main requirements for the structure are:

- shielding
- radioactive containment
- interface capabilities with arriving transport casks
- remote viewing, handling, and retrieval capabilities on the containers
- monitoring of environment within and leaving the vault
- interface capabilities with the departing casks.

The structure also has to provide a safe working environment for operating personnel, ventilation of storage area, removal of decay heat and any gas released by radiolysis of moisture in containers or corrosion, and strength to withstand anticipated static and dynamic loadings on the structure without violation of environmental limitations. Among such loads could be wind, snow, tornado, and seismic forces.

Since the hulls vault can contain significant quantities of undissolved fuel in the cladding wastes, the area may constitute one requiring special safeguards considerations. If so, certain of the Series 5 Regulatory Guides and Title 10 Code of Federal Regulations Part 73 would apply to the design considerations.

18.1.3 Monitoring of Container Corrosion

At the NFS burial ground, as is the case for burial grounds in general, the containers are not relied upon as the primary barrier to the environment. Rather the soil surrounding the drums is the real primary barrier. Section 24.1 presents a more complete discussion of burial grounds.

At the BNFP plant, an engineered, above-ground system is provided for interim storage of hulls. Surrounding the hulls container is a galvanized steel caisson and an earth fill. Beneath the caisson are two compacted layers of clay separated by a gravel layer. Any transport of radioactivity from corrosion of the hulls container which penetrates through the caisson and into the upper clay layer would tend to be detected. Periodic monitoring for liquids (neutron sensitive probes) or radioactivity (gamma sensitive probes) is accomplished by checking lateral monitoring tubes located at the top of the upper clay layer beneath each caisson. A concrete monolith acts as a cover over the caisson berm to exclude entrance of rainwater into soils surrounding the caissons. To verify integrity of the monolith, vertical monitoring tubes in the berm would detect an increase in moisture content of soils before the moisture could descend to the elevation of the hulls containers.

For early detection of potential water accumulation in the caissons which could cause corrosion and eventually transport of radioactivity, each caisson has a monitoring tube extending to the inside, bottom of the caisson. The rate of corrosion from inside the containers should be insignificant, as decay heat will dry the cladding hulls soon after container loading. Corrosion from outside the stainless steel containers should also be insignificant, as long as water does not stand inside the caisson. The container is designed for a 20-year life in the anticipated caisson environment.

In future reprocessing plants, interim storage in a shielded structure might be the choice for providing adequate containment, monitoring, and recovery for treatment. However, if the plants use interim earth storage schemes, they should address the following corrosion considerations: 1) how to provide a moisture barrier to rain and surface waters; 2) how to monitor for water intrusion; and 3) how to detect breaching of a container.

In the unlikely case that water penetrates the hulls container or the rinse water is not fully evaporated before sealing the container, significant pressurization could occur at the maximum expected design temperature of about 350°C within the container. Venting and sealing of the container should be carefully considered because premature sealing is undesirable. However, if the container has a continuous vent, this opening is a potential access point for extraneous water to enter or radioactive materials to escape. Extraneous

water entering the container early in its storage life would likely be evaporated by the decay heat. Development of sealing designs, seal testing procedures, and an operating basis for scheduling the final seal all require further consideration.

During dumping of the dissolver baskets, entrained fines will attach to the container's surfaces and cling there through the washdown operation. NFS uses a ring of water sprays through which the drum passes, and BNFP uses a high-pressure water wand for decontamination of the container's external surfaces. Despite these attempts to remove surface contamination, some residual contamination will be present on the external surfaces of stored hull containers. If further development work indicates sealing of the containers immediately after filling is safe, then future plants should consider providing a decontamination facility for the hulls container.

18.1.4 Recovery for Treatment

Recovery of waste containers from shielded structures (vaults) should be simply a reversed sequence of the loading operations since the integrity of the container should not have changed. By control of the relative humidity within the vault, external corrosion is minimized. In either an NFS or BNFP vault, individual containers could be either transferred directly to an adjacent treatment facility, or loaded into a licensed cask for offsite shipment. If external contamination on the container is above acceptable levels for receipt at the treatment or disposal site, an intermediate decontamination or overpackaging facility would be necessary. Some development may be needed for loading fixtures, decontamination of container surfaces, and provision of an overpack for contaminated containers.

Preliminary design for a single container cask for offsite transport of BNFP hulls indicates that a compound shield of 2.5 cm steel, 18 cm lead, and 5 cm steel would adequately attenuate the radiation. The cask weight with auxiliaries would be about 50,000 kg. This study used a very conservative value for cobalt contamination of Zircaloy of 0.02%. The actual required lead shielding should be about 1.5 cm less than given above since the real concentration of ^{60}Co seems to be only 0.0005%. The limiting fuel exposure history for the cask design was 40,000 MWd/MTU at a specific power of 35 kW/kg, and a decay time of 6 months. An assumed concentration of undissolved fuel in the hulls of 0.5% of that charged to the dissolver is added to the source strength for shielding design purposes.

To recover hulls buried at the NFS facility would be extremely difficult even when documentation shows the location of the deep columns excavated in the soil. Essentially, the reprocessor would have to construct remotely operable excavation equipment to remove the clay above and between these storage

columns. The handling system would have to remove intact drums, gather up bulk wastes from breached drums, and suitably package bulk materials and contaminated soils for future treatment or disposal.

The next step toward recovery would be removal of surface contamination on the hulls package or recanning of the primary container into a secondary container. In either case, the transition from a dirty environment to a clean environment requires development of special facilities, fixtures, and equipment. The shielded cask used to transport the decontaminated drums would have to interface with a load-in fixture at the local or offsite treatment facilities. The cask and fixtures would require design and some development effort. To keep all options open, if use of offsite treatment facilities is contemplated, the cask would probably have to provide for top or bottom unloading capabilities.

A hulls transport cask for offsite shipment of cladding might require a new design concept.

Recovery of hulls containers in preparation for treatment at BNFP requires generally the reverse operation of the loading sequence. The grapple and hoist on the hulls case can remotely couple to the hulls container and draw the container back up into the cask through the bottom doors. If treatment is done onsite, to facilitate transfers from the cask would require design of a load-in fixture. If the treatment facility is offsite, the reprocessor may have to design a licensable cask and design and develop a load-in fixture.

18.1.5 Status of Technology

The technology of containment and burial has been industrially practiced in the past and is currently available if close attention is paid to retrievability. Because burning of waste would likely break the container which must be recovered, pyrophoric material should not be stored. There are some questions regarding container venting, integrity, and surface decontamination that require explicit answers.

1. When can primary containment be sealed?
2. How do you detect primary container ruptures?
3. What methods will be used to decontaminate the outside of waste containers?
4. What methods can be used to avoid external contamination of waste containers?

18.2 CHARACTERISTICS OF TREATED WASTE CONTAINMENT AND INTERIM STORAGE

Before discussing containers, containment structures, and retrieval for final storage, a description is necessary of the likely forms of treated waste which will require handling and storage. Many combinations of treatment options are possible; however, Figure 18.1 shows most of the more plausible alternatives for treatment of cladding hull wastes prior to intermediate storage. The waste forms produced by these treatments are discussed in Section 7.

The waste forms from further treatment of the primary cladding waste which would have to be contained in a shielded structure would include:

1. fuel clad residues mixed with concrete, bitumen, glass, etc.
2. compacted metals
3. chemical compounds
4. dried chemical residues
5. ingots of metal.

18.2.1 Treated Waste Containers

The container should be of a practical size and configuration for intermediate storage, for transport by rail, truck, or barge, and for terminal handling at final storage. Designs for these containers should consider

FIGURE 18.1. Decision Areas in the Management of Fuel Bundle Residues

removal of decay heat, internal or external corrosion of the container materials in the storage environment, and interaction of radioactivity with moisture inside the container.

To minimize the number of handling devices and storage arrays at the final storage facility, this development work should lead to selection of a few basic configurations for the containers used to package treated wastes. In the earliest reprocessing plants to go on-line, the operators have selected certain containers for untreated waste. However, it should be possible to adapt to almost any configuration and material of construction for containers of the products from an onsite treatment facility.

The final seal on the container, the timing of seal application, and means to verify seal integrity will have to be part of the overall development program on containers for treated waste (see Section 18.1.1).

18.2.2 Earth Storage

Intermediate storage by direct burial in soil is one method of holding the less radioactively contaminated containers of treated wastes. The container materials would be chosen such that only superficial corrosion would occur during storage. The storage area would require sufficient monitoring points to detect container failure. (See Section 24.1.)

Placement of containers of highly radioactive treated wastes in underground caissons is a possible method of storage. This system would facilitate monitoring by providing a catch basin for any leakage from the container. The caisson could also act as a second barrier to exclude movement of water around the container. Generally, the burial scheme should provide two barriers surrounding the contained materials with monitoring capability between the barriers. Either water or air samples taken from the caisson would indicate breaching of the container.

18.2.3 Containment Structures

The structure surrounding the containers of treated cladding wastes may be a concrete vault. The capacity of such a vault will depend on how long the reprocessor plans to provide interim storage prior to shipping the wastes to permanent storage. Some of the requirements of this structure have been discussed in Section 18.1.2. These requirements can be modified to compensate for the changed form of the treated wastes.

The designer may wish to take advantage of the differing radioactive decay rates of fuel bundle hardware if these are segregated in the treatment process. Some components (metals) would require less shielding than others. Sorting might also segregate fissile material, which will require criticality monitoring. (See Section 7.)

18.2.4 Retrievability for Final Storage

Retrieval from earth storage of the containers holding most of the radioactive nuclides may be quite difficult. Radiation fields present even after 10 years decay of unsorted waste will require remote grappling of the waste containers for withdrawal into shielded casks. From casks the containers could go through tests of sealing and container integrity in a remotely operated facility. Decontamination and/or repackaging may follow, prior to offsite shipment.

The retrieval system in a vault can be an entirely remote operation or one with limited remote viewing capability. The interface fixture between the vault area and the cask could have a shielded, lead glass window, interior lighting capability, and a remote manipulator to assist in grappling the container with the hook on the cask hoist.

The decontamination/repackaging facility would have to interface with the onsite cask and the licensed offsite cask.

18.2.5 Status of Technology

The technology is generally available. However, the techniques of recovery and corrosion monitoring in earth storage may need development. The radioactive distribution in the treated waste streams needs quantitative definition. The long-term stability of the treated waste forms need to be determined.

18.2.6 Metal Reuse as Interim Storage

Storage and handling costs of metallic wastes can be reduced if these wastes can serve a useful purpose. Ingots melted from cleaned and segregated materials may be suitable for reuse. The fabrication of metal ingots to useful shapes can be accomplished by a variety of conventional metal working processes such as forging, extrusion, rolling, swaging, drawing, machining, and welding if appropriate account is taken of residual radioactivity.

18.2.6.1 Radioactive Control

All metal residues of the fuel bundles will be radioactive (see Sections 2 and 7) and thus must be controlled. The nuclear industry is an obvious industry to give the responsibility for containing the radioactive metal while making use of the metallic properties. Because of the high capital and remote operational costs of metal working tools it is important that reuse applications be directed to simple shapes. Shapes requiring a single forging or extrusion operation should be considered. Applications such as brackets, baskets, buckets, handles, extensions, stirrers, pedestals, etc., may be found in basins or reprocessing plants. Zircaloy ingots, without fabrication, can provide containment for tritium.

18.2.6.2 Recovery

Metal components will be recycled through the melting furnace and reused, or disposed of as waste from that operation. Tritium-containing Zircaloy ingots will be stored.

18.2.6.3 Waste-Process Interactions

Zircaloy--The Zircaloy is used in the core of an LWR because of its low neutron absorption, corrosion resistance, and mechanical properties. These properties are maintained through rigid control of composition, processing, and fabrication variables. It is not envisioned that remelted cladding hulls could be economically refabricated into fuel cladding. However, other applications appear entirely feasible. For applications requiring ingot fabrication, good control of composition will still be required during hull melting to insure the control of carbon, oxygen, and hydrogen necessary to maintain reasonable mechanical and-corrosion resistant properties. Small amounts of additional metallic impurities can be tolerated.

Zircaloy ingots for tritium absorption can be relatively impure and can contain as high as 10 to 20 wt% of stainless steel and Inconel.

Stainless Steel and Inconel--Reasonable strength, ductility and corrosion resistance will be required for reuse applications. These can be maintained with some degree of latitude in compositional control. Up to 5 wt% zirconium in stainless steel or Inconel probably will be acceptable for some reuse applications. Reuse of stainless-Inconel mixtures is also possible.

18.2.6.4 Status of Technology

The degree to which remelted, irradiated fuel bundle residues can be fabricated into reusable forms is unknown. However, unirradiated clean Zircaloy-4 cladding has been remelted and refabricated into cladding tube demonstrating feasibility. After treatment, the hardness and chemical composition of this material was still within nuclear grade specification.⁽³⁾ Ingots of 85 wt% Zircaloy-2 10 wt% 304 stainless steel and 5 wt% Inconel 718 have been melted^(4,5) and samples tested for hydrogen absorption. The results of these tests indicate that this alloy is about as effective as Zircaloy-2 as a hydrogen absorber.⁽³⁾ Areas that may need further development are:

1. Identify reuse applications.
2. Determine the compositional and impurity limitations for reuse applications.
3. Demonstrate the melting and fabrication processes to be used.
4. Show cost-benefit relationship for reuse.
5. Melt, fabricate, and demonstrate the reuse of components made from chop-leach fuel bundle residues.

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19.0 TANK STORAGE OF HIGH-LEVEL LIQUID WASTE

19.0 TANK STORAGE OF HIGH-LEVEL LIQUID WASTE

19.1 HISTORICAL REVIEW

The need for containment of radioactive effluents from nuclear fuel reprocessing plants was met initially by constructing large underground tanks at the fuel reprocessing site. This basic approach, with considerable improvements gained through operating experience and better technology, is still being used today. A thorough review of the operations, progress, development, and experience in tank storage of high-level liquid waste (HLLW) is available in the open literature. This section presents a broad review of the pertinent features of previous tank storage practices throughout the world together with progress made in the management of liquid tank-stored wastes.

A wide variety of tank designs have been employed to contain radioactive waste. This variety has resulted from the diversity in volumes and compositions of waste to be stored, in environmental conditions surrounding the site, and in engineering judgments. In each case, the tank design used reflected a balanced judgment in which operational safety, long-term integrity, waste retrieval, and cost were controlling factors. At most sites the reprocessing waste is first concentrated either in a specially designed external evaporator or by self-heating in the storage tank. The concentrated waste is then stored either as an acidic solution in stainless steel tanks or as a neutralized solution in carbon steel tanks.

At Hanford and Savannah River, HLLW has been neutralized with caustic, which produces an insoluble "sludge" as a precipitate. The neutralized solution and the insoluble sludge can be stored in carbon steel tanks. The liquid solution is further evaporated to salt cake and stored as such. (The lower heat loads in these ERDA wastes allow storage as solid salt cake and sludge.) The salt cake can readily be removed from the tank by re-dissolving in water. The settled sludges are removed hydraulically with liquid jets.⁽¹⁾ The selection of alkaline storage at Hanford thirty years ago was strongly influenced by the fact that stainless steel in quantities required for large waste tanks was comparatively more expensive and not as readily available as it is today.⁽²⁾ Also, no consideration was given to further waste treatment at that time because no solidification methods had been developed. Since then, advances in fuel reprocessing have produced wastes of lower salt concentration and higher heat content, and waste solidification technologies have been developed. These advances favor interim acidic waste storage as a liquid. Therefore, most subsequent reprocessing sites have stored wastes as liquids in the acidic state.

Tanks used for the storage of radioactive process waste have ranged in size from 40- to 4920-m³ capacity. The steel tanks are normally located below ground and are built into concrete vaults. Although the early tanks were of single-wall construction, later tanks had a secondary containment, either in the form of a second steel wall or a steel lining on the wall of the concrete vault (see Figure 1). All high-level storage tanks for acidic waste and many tanks for alkaline waste have been equipped with cooling coils for removing fission-product decay heat. Agitation has sometimes been used to keep solid particles in suspension and to ensure homogeneity of the solution, thus avoiding hot spots and risks of local corrosion. Systems for condensing vapors; for ventilation; and for monitoring tank level, pressure, specific gravity, and temperature have also been used.

As a guide to the historical development of HLLW tank storage, characteristic features of some individual installations are shown in Table 1 and discussed below. As outlined in Table 1, most recent installations have used acidic waste storage.

19.1.1 Neutralized Waste Storage

19.1.1.1 United States

Neutralized HLLW has been stored in the United States for over thirty years. Reprocessing plants at two ERDA sites are currently storing significant quantities of HLLW at Hanford in Richland, Washington, and Savannah River in Aiken, South Carolina. Almost 300,000 m³ of radioactive solutions and sludges are stored at these ERDA sites.⁽²⁻⁴⁾ The only U.S. commercial reprocessing plant storing HLLW is the facility operated by Nuclear Fuel Services, Inc. (NFS) in West Valley, New York. Approximately 2270 m³ of wastes are stored at the NFS plant.^(3,5)

19.1.1.1.1 ERDA Sites. The reprocessing plants at Hanford and Savannah River have stored HLLW as alkaline slurries since startup. These HLLW storage facilities consist of one of two general types of underground tanks: concrete shells lined with carbon steel, called single-wall tanks, and free-standing carbon steel tanks inside concrete vaults. If the vault is steel lined, the latter type is called a double-wall tank.

There are significant differences between HLLW produced in ERDA operations and that resulting from processing of irradiated power reactor (LWR) fuels.⁽²⁾ The total volume of HLLW produced per ton of fuel reprocessed will be considerably less for power reactor fuels than it has been for ERDA fuels, and the heat content will be much higher. These differences are attributed to the mechanical shearing of LWR fuel elements to expose the fuel (ERDA fuels are exposed by chemically dissolving the fuel cladding), to improvements in the reprocessing flowsheets since the ERDA plants were built, and to increased irradiation exposure levels for power reactor fuels.

Single-Wall Waste Storage Tank -
Steel Lined Concrete Vault

Double-Wall Waste Storage Tank -
Freestanding Steel Tank Inside Concrete
Vault Partially Lined with Steel Pan

Double-Wall Waste Storage Tank -
Freestanding Steel Tank Inside Fully
Steel Lined Concrete Vault

FIGURE 1. Typical Waste Storage Tanks

TABLE 1. Tank Storage of Liquid High Level Waste

Site	Operational Experience, years	Waste Source	Waste Type	Tank Size, m ³	Containment Concept	Waste Cooling	Solids Agitation
Hanford (United States)	~31	Production (of isotopes)	Alkaline	190	Single	No	No
				1890	Single	No	No
				2840	Single	No	No
				3790	Single-Double	No	Yes
Savannah River (United States)	~21	Production	Alkaline	2840	Double	Yes	No
				3900	Double	Yes	No
				4920	Single-Double	Some	No
Nuclear Fuel Services (United States)	~9	Power Reactor & Production	Alkaline	2840	Double	Yes	Yes
			Acidic	60	Double	Yes	No
Idaho National Engineering Laboratory (United States)	~20	Production & Naval Fuels	Acidic	1135	Double	Yes	No
Windscale (United Kingdom)	~23	Power Reactor	Acidic	70	Double	Yes	No
				150	Triple	Yes	Yes
LaHague & Marcoule (France)	~17	Power Reactor & Production	Acidic	60	Double	Yes	Yes
				90	Double	Yes	Yes
Eurochemic (Belgium)	~9	Power Reactor & Production	Acidic	40	Double	Yes	Yes
				210	Double	Yes	Yes
				260	Double	No	(a)
				500	Double	No	(a)
Barnwell Nuclear Fuel Plant (United States)	0	Power Reactor	Acidic	1135	Double	Yes	Yes

a. Not found in literature.

Hanford Plant. The Hanford waste storage complex^(2,6-10) includes 149 single-wall tanks ranging in capacity from 190 to 3790 m³ and three new 3790 m³ double-wall tanks. Four additional 3790 m³ double-wall tanks are now under construction.

The first tanks (single-wall) were built for low-heat wastes. These tanks were vented to the atmosphere through reflux condensers. Instrumentation measured tank temperature and liquid level and monitored the surrounding soil for radioactivity. Later tanks, built to contain wastes with greater heat generating characteristics, were generally the same as those for low-heat waste. However, additional features permitted in-tank concentration by self-boiling and necessary recycle facilities that prevent overconcentration. To prevent settling and superheating of the sludge that precipitates when HLLW solutions are neutralized, many of the tanks were equipped with air-lift circulators. Radioactive contamination in the tank vapor space is contained by maintaining the vapor space under a slightly negative pressure and by filtering the noncondensable gases. Explosive gases (chiefly hydrogen generated by radiolysis) are diluted and discharged. The newest tanks are similarly equipped; in addition, double-wall construction⁽²⁾ was used with stress-relieved primary liners.

The Hanford waste tanks do not have cooling coils,⁽²⁾ therefore, these wastes are being processed for the removal of the long-lived fission products ¹³⁷Cs and ⁹⁰Sr to permit in-tank solidification. After short-lived fission products have decayed several years, this treated waste along with other stored low-heat waste is evaporated to salt cake to reclaim tank space and to render the waste less mobile. Since the inception of in-tank solidification in 1965, Hanford has evaporated more than 208,000 m³ of tank-stored waste.⁽²⁾

Since startup of the Hanford plant, 18 tanks have leaked.^(2,3,4,6,7,10,11) All of these leaks were from tanks of the older single-wall design which is no longer used in new construction. The radioactivity that leaked into the soil has remained 30 m or more above the regional water table. No evidence of radioactivity entering the groundwater has been detected, and therefore no detectable or calculable radiation exposure to the public has resulted or is predicted.^(2,3,11) In each case, the leaking tank has been removed from service after transferring its liquid contents to another tank.⁽²⁾

Savannah River Plant. At Savannah River a different approach was used for storing high-heat wastes.^(2,9,12,13) The original storage vessels were freestanding, carbon steel tanks located inside concrete vaults. The vault bottoms were covered with carbon steel liners that extended approximately 1.5 m up the vault wall. The tanks were equipped with cooling coils that used a closed cooling-water circuit to remove heat from radioactive decay. The primary cooling water transfers its heat via heat exchangers to secondary cooling water. Hence, any leakage of activity into the primary loop would be

confined until it could be removed. Dry air is circulated between the tank and vault to keep the annular space dry. Instrumentation monitors temperature, pressure, tank level, and environmental radiation. Forced ventilation purges the vapor space to maintain radiolytically generated hydrogen concentration at a safe level. The tanks are equipped with demisters and filters to collect entrained contaminants.

Twelve tanks were built between 1951 and 1953. Four more tanks built in 1956 are similar except for a roof support modification and an increase in tank capacity from 2840 to 3900 m³.

The newest tanks^(12,13) include primary and secondary liners (double wall). The primary liner is freestanding and stress relieved. Positive air flow is provided between the liners, and cooling is supplied by replaceable cooling coils.

The HLLW is concentrated in an external evaporator installed in each one of two tank farms.^(2,12) The evaporator concentrates are returned to the tanks where salt cake forms as the wastes are cooled. Low-heat wastes at Savannah River are concentrated to salt cake and stored in uncooled single-wall tanks similar to those at Hanford.

At Savannah River there has been one incident in which waste has leaked through the multiple containment of a waste tank.⁽¹⁴⁾ The leak occurred in one of the original tanks in which the vault liner extended only 1.5 m up the vault wall. Only a small amount of high activity waste escaped containment. Calculations based on the measured groundwater velocities, estimated flow paths, and measured ion exchange capabilities of soil samples indicate that the radionuclides that leaked into the ground will decay before reaching the nearest creek and, thus, will have no effect on the public. Leaks have been detected in the primary liner of seven other double-walled, non-stress-relieved tanks.^(2,12,13) However, the escaping radioactive solutions were contained by the secondary steel liners and did not reach the surrounding soil.

Investigation of the leaks at Savannah River revealed that failures occurred at stressed areas near welds due to stress-corrosion cracking accelerated by nitrate ions. Stress relief of the tanks together with sufficient amounts of nitrite and hydroxyl inhibitors is expected to eliminate this type of cracking.^(2,13,15) As a result of this investigation, the most recent waste tanks built or under construction at both Hanford and Savannah River have a stress-relieved, freestanding carbon steel tank inside a fully steel-lined concrete vault.

19.1.1.1.2 Nuclear Fuel Services. The NFS reprocessing plant was the first commercial operation of its kind in the United States. Construction

began in 1963, and hot operation of the plant started in 1966. Experience gained at ERDA reprocessing facilities during the previous twenty years served as the design basis for the neutralized-waste storage system.⁽¹⁶⁾

Currently there are 2270 m³ of alkaline high-level Purex waste (and 45 m³ of acidic thorium waste) stored at the NFS site.^(10,16) The alkaline waste is contained in a single 2840-m³ tank, similar to the first tanks at Savannah River. A duplicate tank is maintained as a spare. The alkaline storage facility consists of a stress-relieved carbon steel tank contained in a steel-lined concrete vault. Because the waste is not yet sufficiently concentrated to maintain self-boiling, a temporary heater has been installed to permit concentration of the dilute waste until either sustained self-boiling or the storage limits are achieved. A condenser connected directly to the tank allows either total reflux or in-tank concentration by diversion of the condensate to a resin clean-up unit. Each tank is equipped with air-lift circulators to promote heat transfer, to suspend the sludge and circulate the contents, and to purge the radiolytically generated hydrogen. No leaks have occurred in these HLLW tanks.

19.1.2 Acidic Waste Storage

19.1.2.1 United States

19.1.2.1.1 Nuclear Fuel Services. High-level acidic waste from reprocessing of thorium fuel is stored in an underground 57-m³ stainless steel tank contained inside a stainless steel-lined vault.^(10,16) A duplicate tank is maintained as a spare. The waste is kept below 60°C by internal cooling coils to minimize corrosion. The necessary auxiliaries for monitoring the waste are provided. To date no tank failures have been observed for this system.

The expansion program for NFS includes plans to store future HLLW as an acidic solution in stainless steel tanks for five years before solidifying as required by present Federal regulations.⁽¹⁶⁾

19.1.2.1.2 Idaho National Engineering Laboratory. At the ERDA reprocessing plant at the Idaho National Engineering Laboratory (INEL) near Idaho Falls, Idaho, high-level acidic waste is stored for appreciable periods in stainless steel tanks prior to calcination.^(2,10,17) The tank farm includes eleven 1135-m³ Type 304L or 348 stainless steel tanks and four other 115-m³ tanks. The 115-m³ tanks are no longer used. Each 1135-m³ storage tank is contained in an individual concrete vault, which serves as a secondary barrier. Eight of the eleven tanks contain closed loop internal cooling coils to maintain the waste temperatures below 65°C for corrosion control. A ventilation system and instrumentation for measuring and recording tank level and pressure, specific gravity, and temperature are provided. The tanks have been in operation since 1953, and no leak has occurred in any tank to date.

19.1.2.2 United Kingdom

High-level aqueous waste generated at Windscale in Cumberland and at Dounreay in Cathness is stored in the acidic condition to minimize waste volumes.⁽¹⁸⁾ Facilities are provided at both sites for the storage of HLLW so that there is no need to transport such waste. No leaks have occurred in the HLLW tanks in the United Kingdom.

19.1.2.2.1 Dounreay.⁽¹⁹⁾ The Dounreay facilities have been concerned primarily with reprocessing spent fuels from reactor development programs. In 1958 a plant for the reprocessing of fuel from Materials Testing Reactors started operation. In 1961 a second plant began reprocessing high-burnup fuel from the experimental Dounreay Fast Reactor. The high-level acidic waste, totaling about 560 m³, is stored in high-integrity stainless steel tanks inside concrete vaults. Cooling coils remove the relatively low level of heat from radioactive decay. Only a small volume reduction has been possible because of the high salt content of these wastes.

19.1.2.2.2 Windscale. Irradiated fuel has been reprocessed at Windscale since 1952, and the site has become the reprocessing center for the United Kingdom nuclear power program. Highly active aqueous waste from reprocessing operations has been concentrated by evaporation. Concentration factors as high as 150:1 with relatively little solids precipitation have been possible because of the low salt content of the acidic reprocessing waste.⁽²⁰⁾ Storage facilities⁽¹⁸⁻²¹⁾ for the concentrated acidic waste comprise eleven stainless steel storage tanks. Eight of these have a capacity of 70 m³ each, and the three newest each have 150-m³ capacity.

The 70-m³ tanks are of the horizontal cylindrical type and are enclosed in stainless steel-lined concrete vaults. The tanks are fitted with internal cooling coils to maintain the contents at 50 to 60°C but are not furnished with agitation equipment. Auxiliaries for ventilation; evaporation; and monitoring of temperature, pressure, tank level, and activity in the surroundings are provided.

The newer 150-m³ tanks are of the vertical cylindrical type. Each tank has a cooling jacket and leak detectors. The temperature of the tank contents is maintained below 60°C to minimize equipment corrosion. A jet ballast and air-lift circulation system is provided for solids agitation, improved heat transfer, and a controlled rate of evaporation. Heated air can also be passed over the waste surface as an alternative and/or additional evaporation mode. Each tank is contained in a concrete vault. The base and walls of the vault are lined with stainless steel. Auxiliary equipment for ventilation; leak detection; and monitoring temperature, specific gravity, pressure, and activity in the surroundings is provided. Means for transferring the waste are provided. Spare storage capacity is always available to guard against

possible failures. This HLLW storage system has been in operation in the United Kingdom for more than twenty years, and will be continued until a suitable solidification process is available.⁽¹⁹⁾

19.1.2.3 France^(22,23)

High-level aqueous wastes from fuel reprocessing plants at Marcoule and LaHauge are first concentrated by two-stage evaporation (70:1 concentration factor) prior to storage in stainless steel tanks. Acidic storage is used to minimize precipitation of solids and to facilitate subsequent waste treatment. Stainless steel waste tanks with useful capacities of 60 and 90 m³ are used at both sites. Each tank is equipped with pulser tubes for agitation and a system for purging the atmosphere above the surface of the waste liquid. The tank contents are maintained below 60°C by either internal cooling coils alone or internal coils in combination with closed-loop water circulation in a pool in which the tank is immersed. In addition to cooling, the pool also serves as a leak detection system. Later tanks have only internal coils for cooling of the concentrated waste. The normal auxiliaries for monitoring the temperature, pressure, specific gravity, and activity of the cooling water and surrounding soil are provided. No leaks have occurred in these HLLW tanks.

19.1.2.4 Eurochemic

The plant of the European Company for the Chemical Processing of Irradiated Fuels (Eurochemic) in Mol, Belgium, was designed to treat spent fuels from both power and research reactors.⁽²⁴⁾ Since active startup in 1966, a large variety of fuels involving different head-end operations have been treated leading to the generation of a great variety of HLLW effluents. The HLLW from reprocessing operations is concentrated by evaporation prior to storage in high-integrity stainless steel tanks.⁽²⁵⁾ Eurochemic has accumulated and stored on site more than 2000 m³ of high and medium level waste.⁽²⁴⁾

To facilitate future waste treatment, storage of HLLW in acidic form was chosen; this choice, however, necessitated the use of stainless steel equipment. The on-site waste storage complex^(24,25) includes two 40-m³ tanks, two 210-m³ tanks, six 260-m³ tanks, and four newer tanks of 500-m³ capacity. The distribution of waste among the four tank types depends on the heat-producing characteristics of the waste and its chemical composition. The 40- and 210-m³ tanks are equipped with facilities for removing heat from radioactive decay and are therefore used for high-heat, low-enriched and high-enriched uranium waste storage, respectively. The uncooled 260- and 500-m³ tanks are used to store decladding waste and aged HLLW that has decayed sufficiently. A double-wall design was used for the two 40-m³ tanks. The closed-loop cooling water circulates in the space between the two concentric tanks and through internal submerged coils. The two 210-m³ tanks do not have double walls. They are

cooled by internal coils. The tanks are contained in heavily shielded lined vaults; the two 210-m³ tanks in one common cell and the 40-m³ tanks in separate ones. Auxiliaries for transfer, ventilation, and monitoring of tank contents and surroundings are provided. The six 260-m³ tanks and four 500-m³ tanks are also contained in lined concrete vaults and are equipped with similar auxiliary equipment.

No leakage from any of Eurochemic's stainless steel tanks has occurred, and no significant corrosion has been observed. However, spare tanks have been provided for these contingencies.

19.1.3 Current Tank Design Status

Over the past thirty years there has been an effort to improve the management and control of radioactive effluents from nuclear fuel reprocessing plants. Considerable progress has been made through advances in fuel reprocessing and waste treatment technologies and with improvements gained through operating experience. This progressive evolution of waste management technology indicates that for today's fuel reprocessing plants acidic waste storage is preferable.⁽²⁾ Storage in this state permits minimizing waste volume, and, more importantly, retains maximum flexibility for later waste treatment. A detailed technical description of a typical modern waste storage system, as exemplified by the Barnwell Nuclear Fuel Plant of Allied General Nuclear Services near Barnwell, South Carolina, is given in 9.2.4.

19.2 CURRENT TECHNOLOGY

19.2.1 Storage Period

The reprocessor of spent, light water reactor (LWR) fuel has the option of converting the high-level liquid waste (HLLW) to a solid form directly without interim storage⁽²⁶⁾ or storing the waste for an interim period as a liquid. Present regulations limit the storage of these liquid wastes to a period of five years.⁽²⁷⁾

Conversion of these wastes to a solid form immediately upon discharge from the reprocessing plant has the advantage of eliminating large-scale HLLW tank storage facilities and of reducing facility surveillance requirements because the waste would be in a less mobile form. The principal disadvantage of immediate conversion is that the radioactivity level would be at a maximum at this time. The solidification processes would have to be designed for the higher heat loads, and these heat loads would consequently limit the size of solid waste storage containers. Also, large quantities of radioactive ruthenium would still be present in the waste and would have to be collected in the off-gas handling facilities of the waste solidification plant.

If the HLLW is stored in tanks for several years, the heat load is significantly reduced from that at the time of discharge from the fuel reprocessing plant (see Section 2.0). The reduced heat loads will minimize cooling and processing requirements. In this assessment, HLLW is assumed to be stored in underground stainless steel tanks in the acidic state for a period up to, but not to exceed, five years following discharge from the reprocessing plant. This is the approach that is being considered for the Barnwell Nuclear Fuel Plant (BNFP) of Allied General Nuclear Services (AGNS) at Barnwell, South Carolina. This plant will be used as a basis for describing typical process conditions for tank storage of HLLW from power reactor fuel reprocessing.

Although storage of waste in tanks is considered an interim measure prior to solidification, waste can be stored in this manner for as long as necessary. To date corrosion of stainless steel waste tanks has been negligible and limitations on the lifetimes of these tanks have not yet been determined. If waste were to leak through the primary container of a tank, it would be collected in the secondary container. The waste from the leaking tank would be transferred to another tank and the leaking tank would be cleaned and taken out of service.

19.2.2 Design Considerations

The design and operation of pretreatment interim storage of HLLW is based on considerations of: (1) safety, (2) ease of waste retrieval, and (3) costs.

The waste can be contained by multiple barriers designed for minimum dependence on people and devices and by surveillance of these containment barriers. All required functions (such as cooling, etc.) are assured by backup systems to maintain acceptable operating conditions at all times.

Retrieval of the waste from tank storage can be simplified by minimizing the volume of insoluble solids and/or providing suitable agitation in the tank to keep the solids from settling.

The options available in designing an acidic waste storage system should be evaluated in light of not only the cost of the interim storage facility itself but also the costs of subsequent waste solidification and storage processes. Costs can be compared directly only for those options that have equivalent degrees of safety.

19.2.3 Process Operating Conditions

A LWR spent fuel reprocessing plant, such as that of BNFP, may generate up to 4 m³ of unconcentrated HLLW for each metric ton* of uranium (MTU) processed.⁽²⁸⁾ British experience indicates this waste stream may be as high

* Metric ton (MT) = 1000 kilograms.

as $5 \text{ m}^3/\text{MTU}$ processed.⁽¹⁹⁾ Before transferring the waste to an interim tank storage facility, the waste is concentrated to minimize storage requirements and to recover most of the nitric acid; the concentration of acid in the concentrate is controlled at about 2 to 4 moles per liter to minimize corrosion of the storage tank.

The degree of volume reduction is determined by two criteria; minimizing precipitated solids in the storage tanks, and assuring that the fission product decay heat does not exceed the tank cooling capacity. Wastes will be concentrated to $\sim 0.5 \text{ m}^3/\text{MTU}$ for storage in tanks.

19.2.4 Features of HLLW Tank Storage

Figure 2 graphically represents the options available for selecting a system for the interim storage of acidic HLLW. The broken line represents the storage system selected by BNFP. More detailed discussion of selected features of this storage system follows.

19.2.4.1 Tank Size

Optimization of waste tank size is based primarily on waste generation rates with upper limits based upon structural and cost considerations. As indicated in Table 2, acidic waste has been stored in cooled waste tanks with nominal capacities up to 1135 m^3 . Figure 3 depicts the storage tank considered at BNFP for interim storage of acidic HLLW. This tank design is similar to that used in the United Kingdom.⁽²⁰⁾ The stainless steel primary container is 16.5 m in diameter and 7.3 m in height and has a nominal storage capacity of 1135 m^3 . Encompassing the primary container is a stainless steel-lined concrete vault. Details of this design are described in the final Safety Analysis Report.⁽²⁹⁾

19.2.4.2 Heat Removal

The primary use of cooling water in a spent fuel reprocessing plant is to remove the heat produced during radiolytic decay of fission products. The amount of heat to be removed in waste solutions depends upon the fuel burnup and the time elapsed between reactor discharge and fuel reprocessing. At BNFP, the cooling system is designed to remove 10.8 MW of heat per tank and to maintain the waste temperature at 60°C or below. This design tank heat load is equivalent to a full tank (1135 m^3) of waste emitting 9.5 W of heat per liter. As indicated in Table 2, this design heat load is within that previously experienced at other reprocessing plants.

Historically, the decay heat has been removed through internal cooling coils. The capacity of such a cooling system is primarily limited from a practical standpoint by the number of coils that can be installed in the tank without taking up an excessive share of the tank volume. At the design full-tank load at BNFP (10.8 MW), 10.9 km of 2-in. schedule 80 stainless steel

FIGURE 2. Design Options for Pre-Treatment Interim Storage of Acidic HLLW

TABLE 2. Operational Experience in the Storage of High-Level
Liquid Fission Product Wastes-Acidic State

<u>Facility</u>	<u>Maximum Tank Size, m³</u>	<u>Tank Heat Load W/liter</u>	<u>Solids Agitation</u>	<u>Operational Experience, years</u>	<u>Volume in Storage, m³</u>
Barnwell Nuclear Fuel Plant (United States)	1135	9.5 (a)	Yes	0	0
Windscale (United Kingdom)	150	10.0	Yes	23	~500
Eurochemic (Belgium)	40	13.0	(b)	~9	<80
Idaho National Engineering Laboratory (United States)	1135	<0.1	No	~20	~7000
LaHague and Marcoule (France)	90	8	Yes	17	(b)

a. Design.

b. Not found in the literature.

FIGURE 3. Schematic of High-Level Liquid Waste Storage Tank

pipe, representing about 3% of the nominal tank capacity, are required to maintain design waste solutions at 60°C or less. In addition, the condenser on the tank vapor outlet is sized to recycle the condensate back to the waste tank in the event of total or partial loss of coil cooling capacity.

19.2.4.3 Agitation

High-level liquid wastes are mixtures of dissolved metallic nitrate salts and insoluble salts. The concentration of insoluble solids depends upon the irradiation history of the fuel, the separations flowsheet, the degree of volume reduction of waste prior to receipt in waste tanks, and the subsequent volume reduction during interim storage. High-level acidic wastes will contain some insolubles⁽³⁾ and, unless means are provided for adequate agitation, local "hot" spots can occur at points where solids accumulate and cause accelerated corrosion of the primary container.

The design of the BNFP waste tank (Figure 3) provides for continuous agitation by air-lift circulators for vertical mixing and continual agitation by ballast tanks to sweep the tank bottom and prevent settling of solids. These techniques are used by the United Kingdom and have proven effective.⁽¹⁸⁾ Even if the heat content of the waste is low enough to permit settling of insolubles into a layer on the tank floor, it will ultimately be necessary to fluidize the solids for ease in retrieval for subsequent waste solidification processes.

19.2.4.4 Corrosion

Stainless steel tanks containing acidic HLLW from LWR spent fuel re-processing facilities have never been known to leak. Available corrosion data indicate that stainless steel will have a long service life provided waste temperatures are maintained less than 65°C.

The expected corrosion rate for the waste tanks at BNFP is less than 0.0025 mm/year.⁽²⁹⁾ This rate is based upon a corrosion and testing program conducted by Allied Chemical Corporation Laboratory in Camden, New Jersey. According to corrosion tests made by the British, stainless steel tanks should experience corrosion rates of about 0.0064 mm/year.⁽¹⁸⁾ At the Idaho National Engineering Laboratory, test specimens immersed in acidic waste solutions show very low corrosion rates, on the order of 0.00025 to 0.0025 mm/year.⁽²⁾ According to K. J. Schneider,⁽³⁾ corrosion rates of 304L or 347 stainless steel tanks by acidic wastes are 0.005 to 0.013 mm/year at 60°C but increase rapidly with increasing temperature to about 1.3 mm/year at temperatures near boiling.

19.2.4.5 Off-Gas Management

Off-gas from a HLLW tank consists primarily of saturated air contributed by the air-lift circulators and ballast tanks plus any hydrogen that may be produced by the radiolytic decomposition of water.⁽²⁹⁾ The hydrogen content

is maintained at less than the lower explosive limit of H_2 in air (4.1%)⁽³⁰⁾ by controlling the amount of air to the circulators and/or ballast tanks. As shown in Figure 3, the exhaust air passes successively through a condenser, knock-out drum, superheater, iodine absorber, and high-efficiency filter before being monitored for radioactivity prior to discharge via a stack.

19.2.4.6 Safety

Location of the primary waste container within additional engineered barriers decreases the probability for leakage of waste to the environment. An additional barrier can be any structure that is impervious to the flow of waste, such as a second metal container or a metal-lined concrete vault. In addition to containing any waste that might be lost from the primary container, the space between the primary and secondary containers can be monitored continuously to determine the condition of the primary container. At BNFP, collecting channels are installed behind all the welds in the secondary container, and a remote monitoring space is provided under the floor liner.⁽²⁹⁾ This allows assessment of the integrity of the welds in the secondary container at any time as well as detection of any waste that might pass the secondary.

19.2.5 Non-High Level Liquid Waste

In addition to high-level liquid waste (HLLW), a spent fuel reprocessing facility will generate a variety of other liquid waste streams which contain much lower concentrations of radioactivity than that in HLLW. For example, intermediate-level liquid waste (ILLW) will be expected to contain approximately 0.0001 as much radioactivity as does HLLW per unit volume, based on both curie and heat content. Typical ILLW streams include concentrated overheads from HLLW evaporation, solvent wash solutions, and equipment decontamination wastes, laboratory waste, etc.

The volume of the ILLW stream is estimated to be approximately 0.1 to 0.3 m³/MTU processed. The ILLW will be stored in the acidic state in 1135-m³ stainless steel tanks identical in construction and containment features to those previously described for interim HLLW storage. However, because of the lower heat content of this waste, removal of radioactive decay heat is not a requirement to maintain waste temperatures below the design level of 60°C. Accordingly, the ILLW tanks will not contain cooling coils. An agitation system is provided to minimize settling of solids.

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20.0 INTERIM STORAGE OF SOLID NON-HIGH-LEVEL WASTES

20.0 INTERIM STORAGE OF SOLID NON-HIGH-LEVEL WASTES

Interim storage methods are not particularly sensitive to the sources of the wastes to be stored. That is, interim storage can take place either before or after one of the treatment technologies described in other portions of this document. Solid non-high-level wastes will be taken to include those wastes that have been solidified or otherwise immobilized. In some cases it is probable that the wastes will receive no further treatment after packing for interim storage and before delivery to the final storage repository.

Current policies and regulations for ERDA wastes⁽¹⁾ require interim storage for only those wastes contaminated with significant quantities of transuranic nuclides. At this time, there is no identified need to store radioactive wastes not contaminated with transuranic activity. However, the storage alternatives described in this section can be applied regardless of whether or not transuranic contamination is present. For these reasons, this section addresses storage of transuranic wastes, the only component of non-high-level solid wastes requiring storage. Should it become desirable to store nontransuranic radioactive wastes, these storage methods could be applied directly to those wastes without adversely changing any of the environmental or safety implications arising from interim storage.

20.1 BASIC CONSIDERATIONS FOR INTERIM STORAGE LOCATION ALTERNATIVES

The most practical and effective choice among the many basic interim storage alternatives is to minimize the number of packaging, handling, and transportation steps between generation of the waste and final storage.

Only two reasons are apparent for not choosing to place the wastes into their final packages and send them directly to terminal storage. Either the terminal storage site and/or methodology does not exist, or substantial economies in radiation exposure, resource commitment, or actual costs can be obtained by delaying final storage or implementing additional waste handling procedures. Since neither the final storage site nor methodology have yet been selected, interim storage of transuranic wastes is the most reasonable alternative.

Alternatives include a single, national interim storage site, multiple regional interim storage sites spaced across the country, or many interim storage sites located at several of the wastes generator's plant sites. Once the location or locations have been selected, choices relating to actual storage methods and packaging can be made, based in large measure on the specific environmental factors at the candidate site.

Twenty years for the interim storage time interval has been selected as the basis for this document. This length of time should prove to be long enough to select adequate final storage and/or treatment methods. Of course, this time interval could be changed as circumstances warrant, but 20 years is representative of the length of time that wastes may have to be stored.

If it is possible to select the packaging requirements for final storage before the site is selected, it should be done. This would allow placing the wastes for interim storage directly in packages acceptable for final storage, thereby eliminating the necessity of repackaging. In any case, the goal of establishing the final storage site and method prior to the designed end of the interim storage life should not be compromised. Rehandling and restorage of initially stored wastes after the interim storage interval appear, at this time, to be inefficient and undesirable.

Another operating sequence is the decision to treat wastes for volume reduction and/or resource reclamation. If treatment is decided upon, then interim storage can be either before or after treatment. The advantage of treating the wastes before storage is that the waste volume to be transported and/or stored may be at a minimum. The disadvantage is that it may not be economical for individual waste generators to all have waste treatment facilities, in which case the wastes will have to be packaged for transportation to a treatment plant. Transportation packages become committed as waste themselves and might qualify as storage containers, perhaps making it advantageous to construct a treatment plant at the storage site and treat the wastes after storage, prior to shipment to the final storage site. The trade-offs involved in this timing decision cannot be adequately assessed until decisions are made regarding interim storage locations and treatment options.

A mixed strategy in interim storage choices may be the best alternative. For instance, it may be best to have a single, national interim storage site for some components of the wastes, and interim storage at the waste generator's site for the balance. Such decisions depend largely on trade-offs between the volumes and types of waste involved, transportation alternatives, specific environmental factors, and political considerations at the waste generator's location.

20.2 DEFINITIONS AND ASSUMPTIONS

To describe the technical alternatives for interim storage of low- and intermediate-level solid transuranic wastes, certain definitions and assumptions must be selected. These definitions and assumptions concern:

- the waste's physical form
- storage locations and intervals
- container requirements
- handling and repository limitations
- possible future waste treatment plans.

For the purposes of this section, the following definitions and assumptions were used. They are based on current ERDA internal policies and requirements⁽¹⁾ and proposed criteria.⁽²⁾ These definitions and assumptions should not be construed as actual requirements that will be imposed and it should be noted that the factors involved in these definitions and assumptions are themselves alternatives, and may be changed.

The definitions and assumptions on which the alternatives described herein are based follow:

1. The waste categories used in this section are defined below:
 - a. Low-Level Transuranic Waste - Wastes contaminated with transuranic nuclides to levels greater than the proposed level of 10 nCi of alpha activity per gram of waste, requiring no shielding for personnel protection, with radiation levels less than 500 mR/hr measured at the package surface.
 - b. Intermediate-Level Transuranic Waste - Wastes contaminated with transuranic nuclides to levels greater than the proposed level of 10 nCi of alpha activity per gram of waste that do require shielding for compliance with occupational radiation exposure limits, but do not require forced cooling.
 - c. The packaged waste containers will not require highly engineered interim storage facilities which provide forced cooling to remove decay heat from the waste containers.
2. The waste's physical form for interim storage will meet the following specifications⁽³⁾:
 - a. Free liquids will be absent. All liquids must be solidified or immobilized before transport to interim storage.
 - b. Untreated pyrophoric or explosive materials will not be permitted.
 - c. Materials capable of spontaneous combustion or production of internal container pressure through chemical or bacteriological reactions will not be permitted.
 - d. Highly toxic or other materials not usually associated with the fuel cycle will be identified to aid in future handling or treatment operations.

- e. Fissile materials will be limited to assure criticality safety during storage.
 - f. Quantities of radioactive materials will be limited to prevent unacceptable temperature rises from decay heat during storage.
 - g. Transuranic materials will be limited to prevent production of unacceptable quantities of alpha radiolysis products during storage.
 - h. Radiation dose rates on packages of low-level transuranic waste (considering in-growth of daughter products) will not exceed predetermined limitations during the storage interval.
3. The storage method will typically be required to comply with the following types of criteria:
- a. Combustible and compactible fractions of the wastes will be packaged separately and segregated from other waste types.
 - b. Package labeling will provide for positive waste identification during the storage interval.
 - c. External package surfaces should arrive at the storage site reasonably free (as specified in applicable regulations) from smearable contamination. Contamination-free is taken to mean the condition of the waste package outer surfaces in which the removable (nonfixed) radioactive contamination will not cause significant radiation exposures to operations personnel.
 - d. Waste containers will be readily retrievable in a contamination-free condition upon termination of interim storage.
 - e. Volume reduction technology, where practicable, will be applied to the waste prior to interim storage.

Interim storage methodologies are basically dependent on these factors:

- packaging of the wastes
- relative location of waste generator and storage area
- transportation requirements
- physical storage method and local environment.

These factors are interrelated. That is, packaging requirements are a function of the waste form, transportation needs, storage method, and the physical environment at the storage area. The storage method is dependent on the type of packages selected. Because of these interrelationships, this section will address generically only those factors important to generalized storage methods.

20.3 ALTERNATIVES FOR PACKAGING OF LOW- AND INTERMEDIATE-LEVEL WASTES

The basic objectives of packaging are to confine the wastes, minimizing the likelihood of leakage of the radioactive materials, and to minimize human radiation exposures. To accomplish these objectives, the following packaging criteria are assumed.

20.3.1 Packaging Criteria

The general packaging criteria established for this document are listed below.

1. All packaging will provide at least two independent barriers designed to contain the radioactive materials throughout handling, transport, and storage.
2. The storage containers will be small enough in volume, light enough in weight, and of some uniform shape so that they are easily handled by conventional materials handling equipment and are compatible with estimates of future treatment facility or final storage size capacities.
3. The storage containers will exhibit good volumetric efficiency in terms of containment of the waste and storage of the containers.
4. All storage containers will meet reliability requirements to assure containment integrity for a minimum of 20 years.
5. The containers should be watertight and capable of resisting the storage atmosphere.
6. The outer container will have sufficient structural strength to withstand stacking, dropping, and penetration tests; meet handling criteria for storage and retrieval operations; and meet shipping criteria for transportation to and from the interim storage site without requiring repackaging of the wastes.
7. The outer container will be capable of being sealed in such a manner that it can be easily determined if the package has been opened.
8. If the containers are constructed from materials subject to corrosion, its surfaces will be covered with a corrosion inhibiting coating.

In addition to the foregoing list, packaging criteria for intermediate-level transuranic wastes will probably also include the following types of items:

1. Sufficient shielding will be provided in either the waste container or in a shielded overpack or other protective system to preclude radiation exposures above occupational limits.
2. All unshielded containers will be compatible with available shielded overpacks or shipping casks.
3. Reusable shielded overpacks and shipping casks will be easy to decontaminate.
4. No storage container for these wastes will require forced cooling.

20.3.2 Low- and Intermediate-Level Waste Containers

The general scheme for the packaging of these wastes is illustrated in Figure 20.1. The waste, packaged in an inner and an outer container, may be placed into temporary onsite storage or may be loaded directly into an approved shipping container and delivered directly to the interim or final storage site.

Because the outer container provides structural support, the inner container would not have to meet as rigorous structural strength requirements, but must be watertight. Some types of plastic can be ideal for inner containers, like the polyethylene liner currently used inside of the DOT 17C drum.

Some form of encapsulation of the waste in a stable solid matrix also may be an acceptable alternative to an inner container. This may be the case after final waste volume reduction and treatment. Encapsulation can only be used effectively on certain waste forms that have been specially treated or will require no further treatment before final storage. Encapsulation is the physical and/or chemical binding of the waste into a stable solid matrix. Glass-ceramic encapsulation is an example. Organic or inorganic polymerization can also be used for encapsulation. Direct production of an encapsulated waste form from a treatment method such as slagging is a possibility. If the matrix is water leachable, however, an inner container may still be necessary. Encapsulation seems to provide good containment, but relative costs will be difficult to obtain until some experience is obtained.

20.3.2.1 Shielded Containers for Intermediate-Level Wastes

The use of containers which incorporate shielding material will generally be limited due to their higher costs. Shielding is not

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FIGURE 20.1. Generalized Waste Packaging Scheme

efficiently used when stored as part of the container. Shielded containers can be used effectively and economically only where small quantities of intermediate-level transuranic wastes are generated and where pretreatment storage is at the waste generator's site.

The shielded outer containers are of two generic types: (a) a container cast from the shielding material and (b) a container in which the shielding is added to the container, allowing variable shielding thicknesses dependent on the radiation level of the waste.

The first type of outer container can be either a cast lead or a cast concrete vault. Either can be made thick enough to supply adequate shielding. These portable vaults would be designed for easy handling with present equipment. They should be easily opened and closed for loading and unloading. Since these vaults would be mainly used where small amounts of wastes are produced, their volumetric efficiency is not of primary importance. The vaults could be coated with a strippable paint to aid in their decontamination.

The second type of container, with add-on shielding, would be useable in the same circumstances as the cast containers, i.e., small waste volume and onsite interim storage.

Two different configurations for this type of shielded container are envisioned. The first would be an inner package placed within a container that has a removable sleeve shielding. The thickness of the shielding could be varied by adding or removing shielding annuli. The outer container in which the sleeve fits would be designed to meet the requirement of the outer package. The second configuration would be a double-contained package shielded by an exterior sleeve. The shielding sleeve may be cylindrical or cubical and would be lowered over the waste package. Additional shielding for the top and bottom of the container would be included as required. Another variation of packaging with add-on shielding is shielding that is incorporated into the system by filling the space between inner and outer containers with molten lead or concrete. This type of container would be expensive, but offers excellent reliability for insuring containment and shielding of the waste.

20.3.2.2 Unshielded Containers for Intermediate-Level Wastes

Unshielded containers appear to provide advantages in packaging the intermediate-level transuranic wastes for several reasons:

- they should be the least expensive to manufacture
- shielded transfer overpacks or casks are now readily available
- the unshielded packages of waste may be more easily handled
- the storage space will be utilized more efficiently without the shielding with each package

- shielding placed around an entire stack of containers is more efficient than placing shielding around indicated containers
- transportation requirements will be simplified if shipping only in approved shipment overpacks or casks.

Unshielded containers can be developed from combinations of the inner and outer containers discussed. The choice of inner and outer containers would depend on the waste form and the extent of package reliability required. In any case, these containers will be designed for ease in waste packaging, transportation, and storage.

A large number of different combinations of containers may be usable for interim storage of transuranic wastes. Materials other than those listed may also be practical because packaging criteria are based on strength and integrity, not on composition.

20.3.3 Summary of Currently Used Packaging

Since 1970, transuranic wastes produced in facilities under ERDA (formerly AEC) control have been subject to requirements for segregation from other types of waste and interim storage as opposed to burial. ERDA Manual 0511⁽¹⁾ currently provides the framework of requirements for this transuranic waste. The information presented in Table 20.1, provided by the ERDA facilities at which interim storage presently takes place, is a brief summary of transuranic waste containers now in use for interim storage.

20.3.4 Transportation of Interim Storage Packages

Shipment of transuranic waste containers will take place in accordance with regulations in effect at the time. Currently, transportation packaging regulations are specified in ERDA, NRC, and DOT rules published as Title 10 CFR 71 and Title 49 CFR.

To meet transportation requirements, two approaches may be taken. The first is to qualify the storage container as an approved transportation package. This requires that each type of storage container pass tests designed to simulate possible transportation and handling accidents. The second approach is to place the storage containers within sturdy overpacks or protective casks which will pass the required tests or regulations.

The first approach, while possible in principle, will generally be too expensive for intermediate-level transuranic waste shipments on a large scale. The second approach is currently in use by ERDA waste shippers, and allows more economical construction of waste containers.

The actual transportation of waste volumes is discussed in more detail in Section 22.0 of this document. However, it may be noted here that

TABLE 20.1. Interim Storage Containers

	Low Level						Intermediate Level		
	Steel Drums	Steel Boxes	Wooden Boxes	Plastic Boxes	Plastic Drums	Other	Unshielded	Shielded	Concrete Vaults
							Steel Drums	Metal Cans	
In use at:									
INEL ^(a)	yes	yes	yes				yes		yes
Hanford ^(b)	yes	yes	yes					yes	
LASL ^(c)	yes		yes					yes	
OR ^(d)	yes							yes	yes
SR ^(e)	yes			yes		yes			yes
Under development at:									
INEL ^(a)				yes			yes	yes	
Hanford ^(b)									
LASL ^(c)					yes	yes			
OR ^(d)									
SR ^(e)									

a. IDO-10051(76); Macbeth & Hickman, "ITSA", Nuc. Tech., Dec. 74; WASH-1536 (To be published).

b. WASH-1538.

c. 1976 Radioactive Waste Management Site Plans for Albuquerque Operations Office, USERDA.

d. Communication from E. M. King, Oak Ridge National Laboratory, October 14, 1975.

e. WASH-1537 (to be published).

transuranic waste containers can be designed to meet applicable shipping requirements. The transportation of waste to and from interim storage sites can therefore be conducted safely and efficiently.

20.3.5 Reliability and Quality Assurance (R&QA) Implications of Packaging Alternatives

In considering R&QA implications of packaging alternatives, a distinction is made between quality of package design and quality of conformance to established design requirements. The quality of package design must be compared to the criteria, specifications, and requirements of the integrated storage method. If a storage method is selected that provides little or no protection to containers, obviously the containers must be of sufficient integrity to protect their contents unaided for the storage interval. If, however, the storage method provides considerable protection to the containers, then they would not necessarily need to be as durable as unprotected containers.

A high level of design reliability is needed for transuranic waste packages to assure that waste does not leak from containers during shipment or storage. Similarly, a high level of quality control during the manufacture of the containers is required. These aspects of reliability and quality assurance should be verified at the waste packagers' facilities to further assure that the integrity of the interim storage method is not compromised. R&QA programs require documentation and auditability to assure the operators of the interim storage site that the waste generator is in compliance with packaging requirements.

20.4 INTERIM STORAGE METHOD ALTERNATIVES

The major requirements for interim storage method alternatives are the ability of the concepts (a) to limit radiation exposures to man and (b) to provide physical storage confinement of the waste, in conjunction with the waste containers, for a period of at least 20 years. At the end of the interim storage period, the waste containers must be capable of being retrieved and transported to future treatment or final storage sites, preferably without the need for repackaging. In addition, consideration will be given to protection of the waste from accidental intrusion, vandalism, and other adverse acts.

Theoretically, prevention of radiation exposure can be accomplished by providing sufficient shielding around the waste product or by providing a sufficient separation distance between the waste product and man. If adequate shielding is provided, then essentially no separation distance is required. Conversely, if the storage concept provides adequate separation distance, no shielding would be required except during handling and transportation operations. Between these two extremes, combinations consisting of varying degrees of shielding and separation distance are acceptable.

In a similar manner, confinement integrity for the wastes can be accomplished by using containers designed to last through the storage interval with no external protection, in which case little or no shelter from the elements need be provided by the storage facility, or by using a storage facility that provides sufficient environmental protection that a waste container of minimal structural integrity will suffice. Again, between these two extremes, numerous combinations of intermediate conditions are acceptable, as shown in Figure 20.2. In addition, it should be noted that the choice of the storage location is interrelated to the waste packaging requirements and storage methods selected.

Interim storage alternatives for low- and intermediate-level transuranic wastes are classified, for purposes of this document, into outdoor and indoor methods. Outdoor methods are taken to be those for which no self-supporting structure is constructed for storing containers; storage can either be above or below the ground surface for this mode. Indoor methods are those which take place inside buildings which can be located either above or below the ground surface. Descriptions of these alternatives follow.

FIGURE 20.2. Relationship Between Container Integrity and Storage Environment

20.4.1 Outdoor Storage Methods

The waste container employed for transuranic wastes and its ability to withstand 20-year storage at a particular storage site, with or without additional protection, will affect the choice of the storage method. Two storage container categories are possible, namely, those that do not need additional protection and those that need some protection from adverse environmental factors to withstand the 20-year storage life. For each of these container categories, several storage concepts are considered as shown in Figure 20.3. These concepts can be divided into those that take place at the ground surface (above-grade storage) and those that take place below grade. The above grade, outdoor concepts consist of:

- totally exposed surface storage
- surface storage with weather cover
- mounded earth cover over storage containers.

Below-grade concepts include:

- hole-in-ground storage
- trench storage
- cave or mine storage.

FIGURE 20.3. Outdoor Storage Method Schematic

Radiation exposure can be reduced by providing shielding material around the waste containers or by providing a sufficient separation distance between the waste products and personnel. While providing only separation distance for eliminating radiation hazards is theoretically possible, the difficulties in maintaining positive control on all persons who may approach the storage area make selection of this method of controlling exposures unlikely. Both shielded and unshielded containers for intermediate-level transuranic wastes are considered. The basic difference between storage of intermediate- and low-level transuranic wastes is that intermediate-level wastes require additional provisions for shielding to reduce direct radiation exposures to operating personnel.

20.4.1.1 Above-Grade Storage Concepts

For above-grade storage, the use of the totally exposed surface storage is limited exclusively to waste containers that possess 20-year service life without additional protection from environmental factors. Surface storage with a weather cover and the mounded earth cover type storage both provide a degree of weather protection so that waste containers which may have less than 20-year service life capability in a totally exposed environment can be used.

20.4.1.1.1 Totally Exposed Surface Storage. As the name implies, in this concept the waste containers would be stored in the open and exposed to the elements similar to the method commonly used in salvage yards. A barrier would be provided to deny access to unauthorized personnel. In addition, posting of security guards and protective measures may be required. The ground surface may be "as is" or a hard surface such as asphalt or concrete may be provided depending on the storage location, its ground surface characteristics, and the local environment. In general, a suitably prepared hard surface is advantageous in facilitating operation of handling equipment and promoting drainage. As a possible consideration, remote, abandoned airfield landing strips may be used for this storage concept. Storage of low-level wastes and intermediate-level wastes packaged in shielded containers would be accomplished in an identical manner. For storage of intermediate-level wastes in unshielded containers, however, the barrier surrounding the storage area that prevents unauthorized personnel access must be located at a sufficient distance from the stored waste. The precise distance will depend upon the radioactivity level of the waste, the degree of shielding provided by the waste container, and the mass of radioactive wastes placed into storage. Additionally, in storing of wastes in unshielded containers, shielding must be provided to prevent radiation

exposure to the operating personnel. As a consequence, while the basic facility cost is low, the costs associated with shielding required during the handling of intermediate-level transuranic wastes packaged in unshielded containers will be high for this storage method.

The major characteristics of this storage concept are:

- possible placement at any acceptable location
- can accommodate waste containers of any size or shape
- provides easy access for in-storage inspection and/or repair of containers
- requires no development to place into operation
- no protection is provided for the stored waste containers against the elements or against adverse events
- may tend to attract vandalism since the waste containers are readily visible
- provides no confinement for the waste in the event that the waste container fails
- does not inhibit the spread of fire, should a fire occur
- may necessitate use of higher priced containers relative to other methods that afford environmental protection.

20.4.1.1.2 Surface Storage with Weather Cover. This storage concept is identical to the previous concept, except that the stored waste containers would be covered with a removable covering. The covering, made from plastic or other similar materials, would be weighted or otherwise secured around the periphery to prevent it from being blown away. Depending upon the particular material selected and the method for securing the edges, it is anticipated that the covering would have to be replaced one or more times during the 20-year storage life. The characteristics of this storage concept are essentially the same as for the totally exposed storage concept, except that the cover provides some protection for the waste containers against the elements and the cost for the facility and for storage and retrieval operations will be slightly higher. In addition, some development effort for selection and securing of the cover material may be anticipated.

20.4.1.1.3 Mounded Earth Cover Over Storage Containers. This type of storage method has been employed since November 1970 at the INEL for retrievalbe storage of low-level, solid transuranic wastes.⁽⁴⁾ This concept utilizes an earth-covered pad storage method. The method consists of a hard-surfaced pad on which containers are stacked. The array of containers is then covered with a protective layer of earth.

Pads are sloped to prevent accumulation of rain or melted snow before the stack of waste is covered with soil. The slope allows sampling of runoff liquids to detect any radioactivity that may escape from the stored wastes. The hard surface and pad width allow utilization of conventional mechanized unloading and container handling equipment. Containers may be stacked higher in the center of the pad than on the edges to allow sloping of the final soil cover to improve drainage. As stacks are completed across the width of the pad, a final cover, which may include plywood for support, of tough nylon-reinforced polyvinyl film and about 1.0 meter of soil is applied successively over the containers. A minimum of 0.6 meter of soil allows operation of earth moving equipment on top of the stacked containers by spreading the weight over a greater area, and is sufficient to establish a vegetative cover to provide for transpiration of some rainfall.

This storage concept could be used directly for storage of low-level transuranic wastes or intermediate-level transuranic wastes in shielded containers. If the intermediate-level waste is contained in unshielded containers, however, this storage concept becomes less desirable. The primary reason is that operating personnel must work in close proximity to the unshielded waste during storage and retrieval operations. Either remote handling equipment must be developed for this operation or shielding must be provided for the operator of the conventional loading equipment such as the forklift truck. The first approach would be preferred but would mean a substantial cost investment. The second approach appears to present such a high degree of potential radiation exposure that it would be unacceptable. Therefore, from a practical standpoint, this storage concept is generally applicable only to storage of low-level waste or intermediate-level waste packaged in shielded containers.

The characteristics of the mounded earth cover type storage concept are:

- facility cost slightly higher than the totally exposed storage concept
- costs for handling of wastes in storage and retrieval slightly higher than either of the two previously discussed concepts
- can be constructed at most locations
- provides for easy monitoring of the environment surrounding the waste containers to detect radiation leakage
- requires little or no development, being currently in use by ERDA
- when covered with earth and vegetation provides camouflage cover for the waste, reducing vulnerability to vandalism and improving aesthetics

- can inhibit the spread of fire within the storage pad
- inhibits spread of radioactive contamination in the event that a waste container fails
- reduces the severity of the local environment to improve the structural life of the waste container.

20.4.1.1.4 Shielded and Protected Storage. In this concept, closed pipes, constructed from concrete, steel, or some other material, would be located above the ground surface, either vertically or horizontally. Waste containers would be loaded into these pipes for storage. When full, the open end would be covered and sealed.

The pipes and their support mechanism might provide some radiation shielding for the contained waste. There appears to be little advantage in storing low-level or intermediate-level wastes packaged in shielded containers in this manner. This might be a usable concept, however, for storage of intermediate-level wastes that require little shielding and are packaged in unshielded containers. Transfer of unshielded containers from a shielded handling cask into the pipe, or vice versa, can be accomplished without exposing the operator to radiation. Other characteristics include:

- it can be located essentially anywhere
- it inhibits the spread of fire
- it inhibits the spread of contamination in case of a container fracture
- the storage pipe structure and operational costs are relatively high.

20.4.1.2 Below-Grade Storage Concepts

For below-grade storage, all three storage concepts provide a certain amount of environmental protection for the waste container. Therefore, any one of the below-grade storage concepts may be used for waste containers that have 20-year unprotected storage life as well as containers that may need some protection to attain a 20-year storage life.

20.4.1.2.1 Hole-in Ground Storage. In this storage concept, holes are provided in the ground to contain a number of stacked waste containers. Concrete or steel pipes could be used as liners for the holes. The holes could be vertical or placed horizontally in an earth wall. As each hole is filled with waste containers, the opening would be closed off with a lid and covered with earth. This lid serves primarily to inhibit entry of water resulting from rain or melting snow. Depending on the storage location, provisions may be required for drainage of the holes to prevent accumulation

of water at the bottom. This can be accomplished by means of sump-type pumps. This type of concept is currently in use for storage of solid intermediate-level wastes. The characteristics of this concept are similar to those noted for the mounded earth cover concept, except that the costs will be greater, both for site preparation as well as for storage and retrieval operations. Additionally, this concept offers greater protection for the waste container, greater containment of the waste in the event of adverse events, but does not provide versatility to accommodate a wide variety of waste container shapes and sizes, and will require some development effort, especially in the area of handling and retrieval equipment.

Since this storage concept offers excellent radiation shielding, there is little advantage in storing intermediate-level wastes that are packaged in shielded containers in this type of facility.

20.4.1.2.2 Trench Storage. The trench or pit configuration is a variation of the hole-in-ground concept, wherein the waste containers would be stored below ground level. The size and shape of the trench or pit would be designed to accommodate the waste containers as well as handling equipment. An inclined ramp could be provided for equipment access. Similar to the previous concept, the walls and bottoms of the trenches could be lined with concrete to prevent wall caveins and to facilitate storage and retrieval operations. When a trench is filled, the open end could be closed off with earth and the top could be covered with plywood, plastic sheet, and earth. Again, provisions for drainage may be required, depending on the storage site location and the corrosion resistance characteristics of the waste containers.

This storage concept may be used for storage of low- or intermediate-level transuranic wastes. In general, the trench concept offers cost savings over the hole-in-ground concept in both initial constructions as well as in handling operations of waste storage and retrieval. Additionally, the trench concept offers greater versatility in effective storage of various sizes of containers. However, if the amount to be stored is very small (where only a few waste containers are to be stored), the cost difference between these two below-surface storage concepts may be negligible. Other advantages and disadvantages of the trench concept are substantially the same as noted for the hole-in-ground concept.

20.4.1.2.3 Cave or Mine Storage. Storage of transuranic wastes in caves or abandoned mines is attractive because this type of storage site, which offers excellent protection from the elements, may already exist. Suitably selected caves or mines will provide near constant temperatures and

humidity, zero precipitation, and freedom from wind and flood hazards. Selection of the storage site location is dependent upon the location of suitable existing caves or mines. Physical security of the area may be provided by installation of protective barriers. This represents additional expenses that may not be necessary for other storage method alternatives located in existing secured areas. Another factor to be considered is the fact that one of the candidates for future permanent storage of transuranic wastes is in mines. Therefore, if a decision to change from interim storage to final storage in a mine were made, provisions for handling and transportation of those amounts of transuranic wastes in interim storage in a suitable mine would be obviated.

20.4.2 Indoor Storage Methods

Indoor storage is an alternative for interim retrievable storage of low- and intermediate-level transuranic waste. Characteristics of the indoor storage methods are:

- the ease for monitoring the storage environment and detection of radiation leakage
- the ability to locate the storage facility almost anywhere
- the ability of the storage facility to accommodate waste containers of essentially all sizes and shapes
- the ease of accessibility for in-storage inspection of the containers
- year-round operations
- storage environment control to protect the containers.

The basic disadvantage of the indoor storage methods compared with the protected outdoor concepts is the higher initial construction cost involved. However, it should be noted that some buildings may be found that would be suitable for storage of waste containers that have fulfilled the functions for which the structures were originally built.

The generic types of structures for indoor storage of low-level transuranic wastes and intermediate-level transuranic wastes packaged in shielded containers, consist of reinforced concrete, wood, metal, and coated fabric-type structures. Several typical construction methods are available, namely, precast concrete, cast-in-place concrete, concrete block, metal frame, coated fabric air supported and coated fabric structurally supported buildings.

A brief description of the alternative indoor storage concepts is presented below.

20.4.2.1 Precast Concrete Structure

The use of precast, prestressed concrete panels in construction of low-cost permanent buildings is a well established engineering practice

(Figure 20.4). Individual panels, approximately 2.4 m (8 feet) wide and any reasonable length, containing prestressed, high-tensile steel wire are cast near the cement mixing plant and transported to the construction site after the panels have been cured. Steel plates are imbedded in the concrete on each side so that the panels can be joined as they are erected into position by welding these steel plates together. The roof and walls are formed from the same basic type of panels.

FIGURE 20.4. Precast Concrete Concept

Storage of waste would take place as in routine warehousing operations. The waste would be delivered to the warehouse in appropriate containers, and stacked using conventional material handling equipment. Depending upon the location of the structure, the local environment, and the structural-life integrity of the waste containers, the atmosphere inside the building could be controlled within temperature and humidity limits. The building could be filled with an inert gas, if desired, and sealed after the building is filled with waste. Air sampling capabilities may be used for monitoring. This type of building is exceptionally strong and relatively low in cost. It is also adaptable to modular construction so that additional units can be added by utilizing common walls for adjacent building units.

From an operational safety standpoint, the precast concrete panel building is inherently safe. The precast concrete building is nearly as fire resistant as a building can be made. The supporting steel is imbedded in the concrete so that it is well insulated against heat and will not collapse, as sometime happens during fires in buildings with exposed structural steel. However, a fire protection system could easily be incorporated into the building design if desired. The building is normally designed to withstand over 146 kg/m^2 (30 psf) snow loading and over 45 m/s (100-mile-per-hour) wind velocity. Heavy rain storms have negligible effect on this building as long as proper drainage is provided. Portable ventilation equipment can be used and moved to additional units.

The concrete construction provides protection for the stored waste against vandalism and accidental access by unauthorized personnel. In addition, the building serves to confine the spread of nuclear contamination in the event of container damage.

At the conclusion of the interim storage period, the building can be cleaned up and used for other purposes. The internal surfaces could be coated with an appropriate paint or resin to seal cracks and pores at the time of construction, to facilitate cleanup. The building could be dismantled and the area returned to unrestricted public use. The dismantled building may have some residual salvage value.

20.4.2.2 Cast-in-Place Concrete Structure

This type of construction is similar to that shown in Figure 20.4, except the walls are cast in place. This is the type of structure generally used for nuclear reactors or processing buildings when massive concrete biological shielding is needed. However, for low-level waste such shielding is not needed and this type of construction would be more expensive than the other types of concrete buildings considered. The roof for this type of building might be constructed of corrugated steel or precast panels as described in the preceding section. Constructing a cast-in-place roof on a storage building of this type would probably be unnecessarily expensive.

The cast-in-place concrete structure would exhibit the same operating characteristics described for the precast building. Fire resistance is excellent. Manpower commitment and land useage would be the same. Basically, the operations inside of this type of building would be the same as in the precast concrete panel structure.

Probably the most negative aspect of this type of construction for an interim (i.e., 20 year) storage building would be the difficulty of dismantling the building after the waste was finally removed. The final

building removal and site cleanup efforts for a cast-in-place concrete building would be much greater than for other types of temporary warehouse structures.

20.4.2.3 Concrete Block Structure

The third basic type of concrete structure is similar to the preceding two, except the walls are laid with cement blocks. The following discussion would also apply to any type of masonry building such as clay brick, cinder block, pumice block, etc. These are considered to be variations of the same general alternative.

The roof on this type of structure is usually supported by exposed steel structural members which constitutes a weakness in case of fire within the building. Although the building itself is fire resistant, a fire within the building contents will frequently cause the steel to over-heat and collapse. This type of building would probably be unsatisfactory for storing combustible wastes without added fire protection.

Storage operations inside a cement block building would be essentially the same as in the other concrete buildings. The type of building would not materially affect the handling operations.

Cement block construction would be somewhat more difficult to seal against surface contamination penetration. At the end of the interim-storage period, the structure could possibly be used for other purposes. However, it is probable that little salvage value could be realized from these structures.

20.4.2.4 Metal Frame

There are several types of metal frame buildings commercially available. Building styles, prices, and quality vary considerably among the various types. These buildings are fire resistant, and can be built to specifications that will meet minimum transuranic waste storage criteria for wind resistance, snow loading, and earthquake resistance at the time of construction.

This type of structure generally has little salvage value unless it could be used for other purposes at the original site after the transuranic waste was removed. Cleanup and dismantling would be relatively easy.

20.4.2.5 Air Support Building

Another type of candidate building to cover interim storage operations is the coated fabric membrane structure supported by internal air pressure. However, the ultraviolet rays of the sun have sufficient degradation effect upon the fabric that a service life of 5 years is

generally considered the expected life of the fabric, with 10 years being the upper limit. For safe 20-year storage, this type of building would have to be replaced periodically.

The air support building shown in Figure 20.5, is usually constructed with a base fabric of nylon or polyester coated with a resinous material such as vinyl or teflon that may or may not contain ultraviolet ray inhibitors. The fabric skin is inflated like a balloon with a slight air pressure. Generally speaking, a positive air pressure between 19 and 32 kg/m^2 (.027 to .045 psi) above the ambient air pressure is sufficient to keep the structure inflated. A large cargo door with an air lock is used for moving large vehicles with freight in or out of the building. Personnel can move freely in or out of a personnel door without the building losing a significant amount of air.

FIGURE 20.5. Air Supported Building Concept

Characteristics of this building type are:

- low initial construction cost and availability
- the building is designed to withstand 45 m/s (100 mi/hr) winds, zone 3 earthquakes, and at least 146 kg/m^2 (30 psf) of snow accumulation
- if used in northern climates, snow and ice must be prevented from accumulating above the 146 kg/m^2 (30 psf) design limit
- the air moving equipment is in constant use and would have a high maintenance cost
- the fabric skin would have less than a 10-year service life
- the positive air pressure inside the building would permit airborne contamination (if any were present) to migrate outside of the building through any building openings.

20.4.2.6 Structural Support Buildings

The structural support fabric building is very similar to the air support building. The cover material is the same type of fabric membrane used for the air support building, but it is supported internally by steel arches rather than air pressure. This structure has the same limitations on fabric life as the air support building but does not require the same amount of heating for snow removal. The structural support building does not depend on blowers to keep it inflated. The initial cost is higher than for an air support building but is less than the more rigid structures discussed previously. This building also can withstand zone 3 type earthquakes. The mode of operation, land commitments, manpower usage, environmental effects, and cleanup and removal efforts would be essentially the same as for the air support building.

20.4.2.7 Structures Placed Below Grade

The various structures discussed previously have all been considered from the standpoint of being constructed above ground level. There is some merit in also considering below-grade or partially buried structures. For example, below-grade storage would provide greater resistance to storms, earthquake, vandalism, or sabotage. However, below-grade excavations are more expensive than placing a similar structure at grade level. One consideration is that, except for the concrete structures, other building materials will be more susceptible to degradation or corrosion if buried below grade. Since the same type of concrete building constructed above grade would meet minimum criteria for earthquakes and other requirements for radioactive waste storage, the only substantial advantages of below-grade storage would be additional protection against adverse events and excellent radiation shielding.

Other below-grade storage concepts, including caves and abandoned mines, are discussed in Section 20.4.1.2.3.

20.4.2.8 Multi-Storied Structures

The use of multi-storied structures is generally appropriate only when the cost of building site land is high. Expensive land is invariably associated with areas of high population density. Many factors would seem to discourage the storage of radioactive waste in urban or highly populated areas. The consequences of a release of radioactive contamination in a highly populated area would be greater than if it occurred in a remote area.

Multi-storied structures are more vulnerable to earthquakes and high winds than are single-storied buildings. Structural strength can be built in to meet earthquake and wind criteria but the cost would be high for a

temporary storage facility. In addition, the storage of heavy containers of waste in a multi-storied structure would require massive support columns which would have to rest on bedrock. This type of structure has been given no further consideration.

20.4.2.9 Structures for Storing Intermediate-Level Transuranic Waste

By definition, intermediate-level transuranic wastes must be provided with adequate radiation shielding or adequate separation distance to prevent radiation exposure to personnel. These wastes can be packaged in either shielded or unshielded containers. If shielded containers are used, they can be stored indoors in any of the structures discussed for interim storage of low-level transuranic wastes. In this application, the building structure again serves primarily to provide a milder environment so that storage and retrieval operations can be carried out more readily on a year-round basis and the structural-life integrity of the waste container would be improved. Although most of the structures will provide some shielding against radiation, it is assumed that since the waste container is shielded, the additional shielding provided by the building structure is not required. The building structures, except for the coated fabric membrane structures, do serve to reduce the possibility of unauthorized entry into the storage area. Similarly, the advantages and disadvantages of each of these types of structures still apply. It is to be emphasized that since the cost of shielded containers is so much greater than that of unshielded containers, the use of shielded containers might be desirable only if the volume of waste to be stored is small.

The high cost of shielded waste containers would generally dictate that the intermediate-level transuranic waste be stored in unshielded containers. In this case, two storage concepts may be considered. In the first concept, the storage building would be constructed to serve as shielding for the waste. Thick-walled, cast-in-place concrete structures would be practical for this application. One concept of this type of storage facility, shown in Figure 20.6, contains a number of storage cells for storing the waste. Waste canisters would be placed in the cells via a bridge crane controlled remotely and can be retrieved in the same manner. The cell size may be as large as needed to accommodate the volume of waste being generated. The concrete walls would be constructed sufficiently thick to provide the degree of protection required for the waste being stored. Waste would be delivered in a shielded cask and would be placed outside the cell labyrinth. The operator would then retire to the control room and remove the waste canister from the cask and place it inside the cell by use of a remotely-operated, TV-camera-monitored crane. The operator would then return and remove the cask.

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FIGURE 20.6. Intermediate-Level Waste Indoor Storage Concept

Another concept for storage of intermediate-level transuranic wastes in unshielded containers is to provide permanent shielded storage locations, such as concrete pipes, holes-in-ground, or trenches within the building. The openings into these shielded storage containers would be designed to mate with the shipping cask so that the waste containers could be transferred from the cask directly into the storage containers without exposing the operating personnel to radiation. This shielded storage container concept may be used in conjunction with any of the building structural concepts described previously, since the building would serve primarily as shelter to facilitate storage and retrieval operations in adverse weather conditions.

20.4.3 Summary of Storage Methods Now in Use

Table 20.2 lists the storage methods now in use or under development by ERDA facilities.

TABLE 20.2. Interim Storage Technologies

	<u>Earth Covered Pads</u>	<u>Pits & Trenches</u>	<u>Pipes in Ground</u>	<u>Concrete Structures</u>	<u>Frame Structures</u>	<u>Air Supported Buildings</u>
<u>In Use At:</u>						
INEL ^(a)	yes			yes		over earth covered pad
Hanford ^(b)	yes	yes	yes			
LASL ^(c)	yes	yes		yes		
OR ^(d)		yes	yes		yes	
SR ^(e)	yes			yes-pipes		
<u>Under Development At</u>						
INEL ^(a)			yes	yes	yes	
Hanford ^(b)						
LASL ^(c)		yes	yes			
OR ^(d)				yes		
SR ^(e)						

- a. IDO-10051(76); Macbeth & Hickman "ITSA" Nuc. Tech. Dec. 74; WASH-1536 (to be published).
b. WASH-1538.
c. 1976 Radioactive Waste Management Site Plans for Albuquerque Operations Office, USERDA.
d. Communication from E. M. King, Oak Ridge National Laboratory, October 14, 1975.
e. WASH-1537 (to be published).

20.5 REPORTING AND RECORDS REQUIREMENTS

Because interim storage implies eventual retrieval of the wastes and shipment to the final storage site, permanent records of the waste placed into interim storage should be kept to aid future retrieval and possible future treatment schemes. Additionally, for purposes of inventory control and nuclear criticality safety, the operators of the interim storage site will need to know the contents of each individual waste package. These requirements impose on the waste generator the need for an in-plant accounting system and waste control method to assure that content criteria are adhered to and that information about waste is available.

A permanent identification label that will be intact and legible for the entire storage interval could be affixed to each waste container. This label may contain a unique identification number which can be cross referenced to the permanent records for that container which give the desired content information.

The inventory and content records can consist of forms developed for this purpose to be filled in as the waste is packaged. The forms would accompany each shipment of waste to the interim storage site. The structure of the forms would be such that the information can be readily inserted into a computer program for data manipulation, storage, and eventual retrieval. The computer program can provide permanency for these records and protection from personnel errors in form preparation through appropriate editing checks.

The following types of information could be provided for each waste container:

- the date sealed
- the permanent, unique identification number for that container
- description and source of waste matrix including such things as hazardous chemicals, compactibility, combustibility, radiation dose rate, and the significant radionuclides present, including more active daughter products which will grow in as storage progresses
- type and volume of the container
- description of packaging material
- gross weight of the waste and container.

For each shipment of waste, the following information might also be provided:

- the date of the shipment
- the permanent, unique identification number for each container in the shipment

- the container types and numbers
- the radiation levels for the shipment
- the total significant radioactive nuclide content of the shipment.

20.6 RETRIEVAL OF PACKAGED WASTE AFTER INTERIM STORAGE

Superficially, retrieval of stored packages is the inverse of storage operations. The differences between retrieval and inverse storage are small, but subtly important to the goals of containing radioactivity and knowing that the radioactivity is contained.

In principle, anything can be retrieved--the radioactivity of an outdoor spill can be gathered up with soil, vegetation, etc., and packaged. All retrieval costs are nonproductive, however, so the question of efficient retrieval of prepackaged wastes should be addressed early. It makes sense to design the storage method and packaging to accomodate effective retrieval operations.

In any case, a closed storage area will be approached, before retrieval begins, as if some packages had failed without being detected. Basically, this involves taking radiation surveys as the storage area is opened and continuously as the packages are removed. Control at this point, particularly in the event that contamination is discovered, is provided well by warehouse type structures and by caves and mines. Soil covered storage areas may be difficult to control upon opening and soil is difficult to survey, especially in poor weather. Near sub-surface storage in vertical pipes is probably intermediate in difficulty of control, but the advantages provided in shielding intermediate-level wastes can logically be traded against subsequent labors of radiation survey. Vertical pipes, however, can only be emptied from one direction. Binding of the containers against the wall could be difficult to overcome and would imprison lower packages. All lifting methods proposed - vacuum, magnetic, and bail - depend on the package having excellent integrity upon retrieval. While this expectation may be well-founded, the failure of a lifting method in the pipe is very difficult to recover from. Hence, vertical-lift retrieval should be avoided unless some special advantage exists, like providing good shielding.

A repackaging capability may be required at each interim storage site before retrieval begins, if only as a precaution for handling failed packages. The sophistication and need of the repackaging capability will depend on (a) the expectations that failed packages actually exist and (b) the problems to be overcome in handling specific kinds of failed packages. The capability would also be needed in case good packages were damaged by incident during retrieval.

20.7 SITE CLEANUP AND DECOMMISSIONING OR REUSE

After completion of the interim storage interval, it will be necessary to reverse the storage procedures, remove all waste containers from storage, and ship them to the final storage or future treatment sites. When the retrieval task is completed, the interim storage site itself must then be dealt with. Individual waste containers are designed so that they will not lose their integrity while in storage or during retrieval and shipment. However, the possibilities of accidents and some failures, allowing spread of some radioactivity, must be recognized.

The difficulty of decontamination efforts will depend on the design of the interim storage facility. Section 15.0 discusses the problems and options available for decontaminating facilities and ultimately decommissioning waste management sites.

REFERENCES

1. ERDA Manual 0511 and WASH-1202(73).
2. D.J. Squires, Waste Form Specification, Packaging Requirements, and Charges for Low-Gamma Transuranic Waste, Draft 11/20/75, ERDA-Hanford.
3. ANSI N14.10.1.
4. 1976 INEL Radioactive Waste Management Plan, IDO-10051, July 1975.

21.0 INTERIM STORAGE OF SOLIDIFIED HIGH-LEVEL WASTE

21.0 INTERIM STORAGE OF SOLIDIFIED HIGH-LEVEL WASTE

According to current regulations,⁽¹⁾ the Fuels Reprocessing Plant (FRP) must solidify the high-level wastes within five years and transfer the solidified waste to a Federal repository within ten years after separation of the fission products from the irradiated fuel. The solidified high-level waste will be in a form such as described in Section 6.0.

It is assumed that the solidified waste will be placed into canisters at the time of solidification, and the canisters will be stored at the FRP in water basins until time of transfer. In Figure 21-1, typical dimensions and characteristics of a canister are noted. The canisters are right cylinders fabricated of stainless steel, Type 300 series. They are closed by welding at the FRP after filling, and after a holding period at high temperature to permit venting of the remaining volatiles. The canister diameter varies between 150 and 610 mm, and the length between 0.6 and 4.5 meters. The heat content of the waste in each canister at time of shipment to the Federal repository is a maximum of 13 kW.

Final geologic storage sites and/or methods are under evaluation and may not be available when waste deliveries begin. Studies have been conducted within the Energy Research and Development Administration (ERDA) to provide a Retrievable Surface Storage Facility (RSSF) for interim storage and isolation of the waste from man and his environment until final geologic storage has been established. The RSSF would be designed to store the solidified high-level waste for at least 100 years, although it is not expected that the waste will need to be held in interim storage for that length of time. It is expected that final geologic storage will be ready in tens of years and at that time the stored waste in the RSSF would be moved to the final geologic storage site. In view of the prospective delays in generation of commercial high-level waste and in view of proposed acceleration of site selection for geological isolation (final storage), ERDA now believes the eventual need for retrievable surface storage will be much less than when the RSSF work was begun in 1972.

The three basic cooling concepts chosen for detailed study were natural circulation air-cooling in sealed storage casks, forced circulation water cooling in basins, and natural circulation air-cooling in vaults.

Location alternatives for interim storage can include: interim storage at a single national site, interim storage at the FRP, interim storage at regional repositories, and interim storage at the final geologic storage site(s). These last three alternatives have not been studied extensively.

The characteristics of the alternative interim storage technologies for solidified high-level waste are discussed in this section. For each of the alternatives a technical description is presented and the length of the

description is proportional to the importance of the alternative and the amount of work performed on the alternative. The status of the technology of each alternative is also presented. This includes discussions on the estimated time of readiness for practical design and the major development program objectives that need to be met for each alternative.

SIZE**150 TO 610 mm DIAMETER****0.6 TO 4.5 m LENGTH****MATERIAL****300 SERIES STAINLESS STEEL****HEAT****<1 TO 13 kw**

FIGURE 21-1. High-Level Waste Canister

21.1 BASIC CRITERIA FOR INTERIM STORAGE FACILITIES

Each interim storage facility alternative will be capable of receiving, processing, and safely storing the solidified high-level radioactive waste from one or more 1500 MTU/yr reprocessing plants, depending upon interim storage facility location. Planned operating methods and design concepts will be based on established and proven technology. Retrieval of the stored waste would be possible at any time during the storage period. The facilities should be designed to permit future decommissioning. Design of the heat removal systems will be such that under no conditions can the solidified high-level waste attain its maximum processing temperature. Constant surveillance of the operations, plant protection, and physical security are to be provided.

The interim storage facilities will be designed to contain and control liquid, solid, and gaseous waste resulting from normal and abnormal operations of the facility and from exposure to natural phenomena. Design of the radiation protection systems will be based upon meeting limits which are as low as is reasonably practicable and are below the limits prescribed in the Code of Federal Regulations, Title 10, Part 20 (10 CFR 20).

21.2 COOLING CONCEPTS FOR INTERIM STORAGE OF SOLIDIFIED HIGH-LEVEL WASTE

Engineering studies of alternative cooling concepts for interim storage of commercial high-level waste have been made.⁽²⁾ The concepts studied include the Sealed Storage Cask Concept (SSCC), Water Basin Concept (WBC), Air-Cooled Vault Concept (ACVC), and a few others which are mainly takeoffs from one of the above listed concepts. Conceptual facility designs and conceptual overpack or storage cask design and fabrication methods for each concept were defined. (See Figure 21-2.)

At the request of the U. S. Atomic Energy Commission (AEC, now ERDA) the Panel on Engineered Storage of the Committee on Radioactive Waste Management of the National Academy of Sciences - National Research Council, analyzed the concepts proposed by the AEC to store solidified high-level radioactive wastes in retrievable surface storage for a period of up to 100 years. The Committee endorsed the conclusions and recommendations of the Panel, from which the following excerpts are quoted: "The Panel has evaluated the available, pertinent information on management of high-level radioactive wastes that result from the operation of nuclear power reactors. It finds that retrievable surface storage can be developed that will (a) ensure public safety, through multiple barriers against dispersal of radionuclides and against exposure of

FIGURE 21-2. Cooling Concept Alternatives for Interim Storage of HLW

persons to radiation; (b) minimize adverse effects on the environment; (c) permit operation within the necessary time scale; and (d) minimize the increase in nuclear power costs."

"The Panel concludes that retrievable surface storage is an acceptable interim stage in a comprehensive system for managing high-level radioactive wastes."

"The Panel has reviewed alternative methods of radioactive waste processing and storage. It concludes that, in principle, high-level radioactive wastes can be stored safely and economically in either sealed storage casks, in a water basin, or in an air-cooled vault. However, the Panel recommends an optimized version of the sealed storage-cask concept."⁽³⁾

21.2.1 Sealed Storage Cask Concept (SSCC) Alternative

INTERIM STORAGE OF SOLIDIFIED HIGH-LEVEL WASTE SEALED STORAGE CASK CONCEPT

FLOW DIAGRAM

Storage Unit Description

A canister of waste is placed inside of a 51 mm/thick carbon steel storage cask. The cask is placed inside of a 0.95 m thick concrete shield. The annulus between the storage cask and the inside face of the concrete shield permits air-cooling of the storage cask by natural convection. Air enters the annulus at the bottom of the storage unit and exits near the top. Storage units are placed above ground in the storage area.

Applicable Wastes

Solidified High-Level Waste.

Process Description

Shipping casks containing canisters of waste are unloaded in a shielded facility, which is remotely operated. Canisters are examined and then decontaminated, if necessary. In a second shielded facility the canisters are placed inside of a storage cask which is then sealed by remote welding. Non-destructive examination of the storage cask closure is performed, and the cask is placed inside of the concrete shield to form the storage unit. The storage unit is then transported to the storage area on a tractor-trailer. A yard gantry crane removes the storage unit from the trailer and places it into position on the ground.

Principal Advantages

A passive system for cooling the waste; protection of the canister by steel and a thick concrete shield; and a minimum amount of heat and radiation interaction between storage units.

Principal Disadvantages

Welding and nondestructive examination must be performed remotely; storage units are large and heavy to transport.

Development Status

No developmental work is required. A demonstration program is in progress to verify the design work.

Process Effluents

Solid nuclear waste is placed into drums for disposal; radioactivity in the gaseous effluents is negligible, water vapor is released from the cooling

tower and from the evaporation pond; liquid effluents are collected in and evaporated from the pond; accumulated salts from the evaporation pond are trucked to a disposal area.

The Federal repository for interim storage of the solidified high-level waste would provide the facilities for safely processing and storing the high-level radioactive waste generated by the nuclear power industry for an interim period. Based on the results of recently completed studies, the air-cooled SSCC is one of the most attractive methods that could be applied. This is the approach recommended by Reference 3. A description of the SSCC conceptual design is presented below.

21.2.1.1 Technical Description

The interim storage facility using the SSCC alternative for cooling the waste would consist of two principal areas. These are the receiving and assembly building and the storage area. In the receiving and assembly building waste canisters are received and placed into storage units. In the storage area the units are stored above ground and are kept under surveillance.

- General

Within a ten-year period following fuel reprocessing, the FRP shall ship the waste canisters to the Federal repository in heavily shielded casks. Upon receipt at the repository, the shipping casks are opened and the waste canisters are inspected for radioactive contamination and damage. Each canister is then encapsulated in a 5 mm thick wall, carbon steel storage cask which is next inserted into a concrete gamma-neutron shield and transferred to an open storage area. The encapsulated waste canisters will remain in the storage area under surveillance until they are transferred to a final geologic storage site. The decay heat is rejected to the atmosphere by air circulation through cooling passages in the storage unit. The repository has facilities and equipment for decontaminating the waste canisters upon receipt, overpacking failed canisters, and cleaning contaminated shipping casks as well as providing service and support functions.

- Receiving and Assembly Building

The receiving and assembly building consists of the following operating areas: receiving area, weld and test area, and storage unit assembly area. (See Figure 21-3.)

Receiving Area

The receiving area provides for the receipt of shipments, shipping cask handling and inspection platforms and equipment, and a shielded receiving cell for shipping cask unloading, and the inspection and the transfer of the waste canister.

FIGURE 21-3 Conceptual Layout - Receiving and Assembly Building

The waste canisters are unloaded from the shipping cask into the heavily shielded remotely operated receiving cell. There the canisters are examined and, if necessary, contamination is removed in the decontamination chamber. If the canisters are thermally hot on arrival, controlled cooling of the canister will be performed before decontamination.

The receiving cell is equipped with manipulators and a remotely operated crane to assist in cell operations. It is also equipped to overpack failed canisters, if necessary, before transfer to the weld and test cell.

Weld and Test Area

The weld and test area consists of the weld and test cell and its accompanying operating and service galleries. The weld and test cell is a shielded facility in which the storage casks, containing the waste canisters, are processed for storage. Remote operations in the cell are conducted by the operators through the use of shielded viewing windows.

The weld and test cell is equipped with manipulators, remotely operated cranes, and powered lifts to assist in cell operations. The process operations conducted in the weld and test cell include: storage cask closure welding, nondestructive examination of the storage cask welded area, stress relief of the storage cask welded area, and storage cask weld removal.

Storage Unit Assembly Area

The storage unit assembly area provides the assembly fixtures and handling equipment required to: receive subassemblies of the storage unit, assemble and inspect the storage unit, transport the storage unit

within the area, and place the storage unit on the tractor-trailer for transport to the storage area.

- Storage Area

After completion of the testing and inspection operations in the weld and test cell, the storage unit is delivered to the storage area by the tractor and trailer from the receiving and assembly building. In the storage area a mobile yard gantry crane is used to remove the storage unit from the trailer and place it on a storage pad. The arrangement of the storage pad is based on consideration of radiation dose rate and dissipated heat. Aisles are provided for the yard gantry.

- Storage Unit

A conceptual storage unit is illustrated in Figure 21-4. The dimensions listed for this storage unit represent maximum dimensions as currently envisioned. The canister is 610 mm OD by 4.5 meters long. The carbon steel cask is 760 mm OD by 660 mm ID by 4.7 meters long. The concrete gamma-neutron shield is 3 meters OD by 1.1 meters ID by 6.3 meters long. The concrete cap is 840 mm thick and is equipped with a removable plug in the center for insertion of the sealed storage cask. The shield is supported by a concrete pad, and the storage unit, including pad, weighs about 118 tonnes.

The sealed storage cask is supported in the middle of the storage unit such that air can flow around it, entering at the bottom and leaving at the top to remove the heat of decay from the waste. The storage units would be placed above ground in a storage area enclosed by fencing. Spacing of the storage units would be selected to permit vehicular traffic needed for maintenance and handling of the storage units, and for temperature and radiation control in the storage area to permit controlled personnel access.

The wall thickness of the carbon steel storage cask is 51 mm. The maximum normal operating temperature of the storage cask is estimated to be 316 °C, and the maximum oxidation rate of the carbon steel at this temperature is estimated at .05 mm per year. The storage cask is designed for a 100-year life, hence 5 mm is the total design corrosion allowance for the storage cask. The additional 45 mm of wall thickness is furnished to permit safe handling of the heavy storage cask and to enable it to safely withstand a drop from the maximum possible height it could attain during operations, should a drop occur.

The upper temperature limit for usefulness of carbon steel in air is 649 °C, and this is the maximum temperature that the storage cask can

FIGURE 21-4. SSCC - Storage Unit

attain during operations as it is limited by storage unit design. Such an occurrence could take place if there was a complete air flow stoppage in the storage unit, such as might happen from the accumulation of dust or dirt at the inlet ports. It is not anticipated that such an occurrence would go undetected for any length of time due to the surveillance program which would be in effect, and which is discussed below. Should a complete air flow stoppage occur, the storage unit temperature would rise. No damage to the storage unit should occur as the design of the storage unit obviates the attainment of waste temperatures above its maximum processing temperature. An air flow stoppage would be detected by the rising temperature of the storage unit and then the storage unit would be returned to the weld and test cell and examined for damage. If needed, the storage cask could then be overpackaged, or the canister could be repackaged in another storage cask.

Facilities and Equipment

The storage area is subdivided into lots and the geometric arrangement of the storage units within the lots is selected to limit the radiation dose to personnel in the storage area to design limits. Spacing of storage units within the lots also takes into account the tractor-

trailer and yard gantry crane movement requirements. Between lots, aisles of sufficient width are provided to facilitate turning of the tractor-trailer into straight line alignment with the aisles within the lots.

The trailer, Figure 21-5, is a high tonnage capacity, low bed style, four-wheel trailer running on earth moving equipment style, pneumatic tires. The yard gantry has a high tonnage capacity, equal to that of the trailer, and is designed for on-board control by an operator.

FIGURE 21-5. Partial View of Storage Area - Conceptual Arrangement

The storage area site is fenced with a high chain link fence and perimeter lighting is provided. Each lot of storage units inside the storage area is also individually fenced with a high chain link fence. The area between the foundation pads and the heavily traveled aisles is treated with a defoliant, graded, and surfaced with crushed rock.

All heavily traveled aisles are provided with adequate amounts of crushed rock to withstand the traffic anticipated in the design of the facility.

Retrievability

The geometric placement of the storage units in the storage area permits retrievability of the storage units through use of the yard gantry crane and the tractor-trailer. The yard gantry crane lifts the storage unit and places it onto the trailer for transport back to the receiving and assembly building. There the storage cask can be removed for inspection purposes, or for packaging prior to transport to the final geological storage site. Repackaging of the canister or overpacking of the storage cask, if required, could be accomplished in the weld and test cell.

Monitoring and Surveillance

A monitoring system is provided for taking routine measurements for the purpose of documenting compliance with existing regulations. The monitoring system consists of fixed monitors, recording continuously, and mobile monitoring performed by radiation protection personnel who make area surveys, wipes, and air sample tests at specific locations on a programmed basis.

A surveillance system is provided for containment verification. This is primarily achieved through observance of the facilities in terms of anticipating problems and detecting a possible release at the storage unit level in time to contain the release within the confines of the site. The storage area on-line surveillance system provides: immediate detection of a radioactive release, positive identification of failed storage units, and possible detection of impending storage unit failures, which would be accomplished with instrumentation other than radioactivity detectors and/or with a nondestructive examination program.

21.2.1.2 Status of the Technology

The SSCC is based on technology which is well known in the nuclear industry. The principal technical features are welding, heat transfer, and radiation shielding design. No difficulties are anticipated in the application of these technologies to the particular needs of the SSCC, even though some of the applications are unique.

- Estimated Time of Readiness for Practical Design

Conceptual design engineering studies have been performed for the SSCC alternative by ARHCO for ERDA in the Retrievable Surface Storage Program. A draft of the design criteria, which would serve as a basis for preliminary engineering design, has been completed. About nine months would be required to complete the design criteria, select an architect-engineer for preliminary engineering design, and start

preliminary engineering design. With the proper approvals, design and construction would be completed so that facility startup could commence within six years after start of preliminary engineering.

- Major Development Program Objectives To Be Met

No major development work is required for the implementation of the SSCC alternative. However, the adaptation of the technologies to the SSCC is unique and it is desirable to demonstrate the concept feasibility and the safety features. Currently, a demonstration program is being conducted by ARHCO for ERDA (see Figure 21-6). The primary objectives of the demonstration program are as follows:

FIGURE 21-6. Storage Unit Demonstration at Hanford

- Construct and operate a nonradioactive storage unit for design verification.
- Demonstrate structural integrity of the sealed storage cask.
- Construct and monitor a storage unit containing high-level waste to obtain operating experience.

This work is in progress and is scheduled to be completed in about two years.

Other work which also may need to be performed includes the demonstration of the storage unit surveillance instrumentation systems. About two to three years will be required for this program.

21.2.1.3 Variant Alternative - Sealed Storage Cask Concept - Medium Wall

A variance in the SSCC is the Sealed Storage Cask Concept With Medium Wall (SSCC-MW). The SSCC-MW is similar to the SSCC described above except that the cask provides more shielding and no outer concrete gamma-neutron shield is used. Overall there is less equivalent shielding and this must be compensated by the use of portable shielding and greater distances in the storage area.

The SSCC and the SSCC-MW represent two extremes in that the wall thickness of the storage casks varies from 51 mm to 203 mm, and the concrete shield thickness varies from 0.95 meter to zero. Variant alternatives are also feasible for storage cask wall and concrete shield thickness in between these. Another variable in design is the allowable radiation in the storage area. In the case of the two extremes, personnel can work in the storage area for limited amounts of time without additional shielding with the SSCC, but for the SSCC-MW, shielded vehicles from which the storage area work is conducted are necessary for personnel.

A description of the SSCC-MW conceptual design is presented below.

● Technical Description

The interim storage facility using the SSCC-MW alternative for cooling the waste would consist of two principal operating areas. These are the receiving and assembly building and the storage area. In the receiving and assembly building waste canisters are received and encapsulated into storage units. In the storage area the storage units are stored above ground and are kept under surveillance for an interim period.

General

Operations in the receiving and assembly building are similar to those for the SSCC, except for the gamma-neutron shield used in the SSCC. In the weld and test cell each canister is encapsulated in a 203 mm-thick wall, carbon steel storage cask to form a storage unit. The storage units are moved to the storage area in a heavily shielded vehicle and stored on cradles in a horizontal manner. The decay heat is rejected to the atmosphere by air circulation around the storage unit (see Figure 21-7).

The design of the SSCC-MW storage unit utilizes a carbon steel storage cask which has a wall thickness of 203 mm in which the canister is

FIGURE 21-7. Sealed Storage Cask Concept-Medium Wall, Storage Unit on Concrete Saddles

placed. The estimated operating temperature for the surface of the storage cask of the SSCC-MW storage unit is 82 °C. Since the storage unit is always exposed during storage, a corrosion rate of 0.25 mm per year is used with a design life of 100 years. Hence, the maximum corrosion allowance is 25 mm. The remaining 178 mm is provided to furnish a rugged wall cask to withstand tornado-generated missiles and the like. The metal also provides a degree of radiation shielding.

Receiving and Assembly Building

The receiving and assembly building of the SSCC-MW is basically the same as that for the SSCC, with the exception that facilities to handle the permanent, concrete gamma-neutron shields for storage units are not provided. Facilities to handle unshielded storage casks are provided. The medium wall storage cask has a wall thickness of 203 mm which is not sufficient for gamma and neutron shielding; therefore, a transfer shield is provided, which furnishes the additional shielding for canister transport within the facility. A shielded loadout corridor is provided for storage unit transfer to the transporter loading station.

Storage Area

A heavily shielded transporter picks up the storage unit at the loading station and carries it to the storage area. Storage units are stored horizontally end-to-end in rows spaced on 6 meter centers and are supported on concrete saddles with crushed rock around the saddles (see Figure 21-8). A shielded storage area utility vehicle is provided to perform maintenance and surveillance functions. The retrievability features of the SSCC-MW and the monitoring and surveillance systems provided are similar to those provided for the SSCC (see Section 21.2.1.1). The radiation dose rates within the storage area are extremely high and shielded personnel carriers must be used to perform any work in the area. Within 610 m of the storage area the radiation dose rates are also very high and personnel shielding is also required for performance of work in this area. Outside of the 610 m distance from the storage area no personnel shielding is required.

FIGURE 21-8. Sealed Storage Cask Concept - Medium Wall, Storage Area

- Status of the Technology

The status of the technology for the SSCC-MW is similar to that discussed for the Air-Cooled Vault Concept (ACVC) (see Section 21.2.3.2.).

21.2.2 Water Basin Concept (WBC) Alternative

INTERIM STORAGE OF SOLIDIFIED HIGH-LEVEL WASTE WATER BASIN CONCEPT

FLOW DIAGRAM

Storage Unit Description

Bare canisters are stored in support racks underwater in a water basin. The basin is constructed of concrete and is lined with stainless steel. Heat is removed from the canisters by the water in the basin. The basin water is cooled by pumping it through a water-to-water heat exchanger. Heat is removed from the secondary cooling stream by a water cooling tower which dissipates the heat to the atmosphere.

Applicable Wastes

Solidified High-Level Waste.

Process Description

Shipping casks containing canisters of waste are unloaded in a shielded facility, which is remotely operated. Canisters are examined, cooled, and decontaminated, if necessary. The canisters are transferred in a shielded cask to a water basin where they are unloaded under water and placed into racks under water for storage.

Principal Advantages

High capacity heat sink available to absorb heat upon loss of normal cooling systems; demonstrated technology in reactor and reprocessing plant fuels storage basins; capable of handling high heat fluxes; water serves as a confinement barrier in the event of container failure; and water provides shielding and permits visual surveillance.

Principal Disadvantages

Potential for container failure from corrosion; high purity water required; decontamination of water is a significant operating problem; active cooling system requires constant operation and attendance.

Development Status

No developmental work is required, but a demonstration program would be desirable to verify design calculations.

Process Effluents

Solid nuclear waste is placed into drums for disposal; radioactivity in the gaseous effluents is negligible, water vapor is released from the cooling

tower and from the evaporation pond; liquid effluents are collected in and evaporated from the pond; accumulated salts from the evaporation pond are trucked to a disposal area.

Water basins have been used successfully for the storage of spent reactor fuel elements for many years in the nuclear industry. The water provides cooling and shielding, and permits the operators to view the underwater operations. The technology can be applied to the storage of canisters of high-level waste in water basins also. This concept can be used by ERDA for interim storage at a Federal repository or by a FRP for interim storage at its plant.

A description of the WBC conceptual design follows.

21.2.2.1 Technical Description

The interim storage facility using the WBC alternative for cooling the waste would consist of two principal areas. These are the receiving building and the storage area. In the receiving building waste canisters are received and placed into a transfer cask for transfer to the storage area. In the storage area the transfer cask is received and unloaded under water. The canisters (see Figure 21-1) are stored under water in basins and are kept under surveillance for an interim period.

- General

Within a ten-year period following fuel reprocessing, the FRP shall ship the waste canisters to the Federal repository in heavily shielded casks. Upon receipt at the repository, the shipping casks are opened and the waste canisters are inspected for radioactive contamination and damage. Each canister is then placed into a heavily shielded cask and the assembly is transferred to the storage area. There the transfer cask is unloaded under water and the canister is placed in a rack and stored under water. The canisters shall remain in the storage area under surveillance until they are transferred to a final geologic storage site. The decay heat is removed from the storage basin water and rejected to the atmosphere by forced draft cooling towers. The repository also has the capability of decontaminating the waste canisters upon receipt, overpacking failed canisters, and cleaning contaminated shipping casks, as well as providing service and support functions.

Receiving Building

The receiving building is a structural steel-framed and reinforced concrete structure consisting of the following areas: shipping cask receiving area, receiving cell, and transfer corridor. (See Figure 21-9.)

FIGURE 21-9. Water Basin Concept - Receiving Area Layout

Shipping Cask Receiving Area

The shipping cask receiving area is served by an overhead crane capable of handling high tonnage shipping casks. Capability is provided to unload shipping casks from either a truck or a rail-road car. Either dry or wet shipments of canisters could be handled; but this design description addresses dry shipments only.

Receiving Cell

The waste canisters are unloaded from the shipping cask into the heavily shielded, remotely operated receiving cell. There the canisters are examined. If the canisters are thermally hot on arrival, controlled cooling of the canister will be performed in the decontamination chamber. Decontamination of the canister, if needed, will be accomplished before transfer to the storage area.

The receiving cell (Figure 21-10) is equipped with manipulators and a remotely operated crane to assist in cell operations. It is also equipped to overpack failed canisters, if necessary, before transfer to the storage area.

Transfer Corridor

The transfer corridor connects the receiving cell with the storage area. Each inspected and cooled waste canister is transported from the receiving cell to the storage area in a water-filled shielded transfer cask mounted on a rail car which runs on rails in the floor of the transfer corridor. When the cask arrives at the water basin it is lowered into the basin and the canister is transferred from the cask to the storage racks.

FIGURE 21-10. Water Basin Concept - Receiving Cell

Storage Area

The waste storage basins are stainless steel-lined concrete cells, 11 meters deep by 5-1/2 meters wide by 11 meters long, each with a nominal capacity of 500 waste canisters stored in double-tiered racks for support. (See Figure 21-11.) The basins are filled with de-mineralized water which cools the waste and provides transparent radiation shielding and a confinement barrier. Each storage area building contains a fixed number of storage basins. The storage area is designed to be built in building modules so that it can be expanded to meet increasing inventory requirements (see Figure 21-12).

Storage Area Building

The water basins are covered by a building which contains an overhead bridge crane, crane supports and services. The building provides personnel comfort and prevents dirt and debris from entering the water basins. Metal covers are provided for the basins to allow free personnel movement. They also provide temperature and humidity control if a canister leaks. Water depth of 4 meters over the waste canisters provides personnel radiation protection under normal conditions, but thick concrete blocks will be used to cover a cell in the event of a canister rupture and gross contamination.

FIGURE 21-11. Water Basin Concept - Canister Storage Basin

FIGURE 21-12. Water Basin Concept - Conceptual Storage Area Expansion

Storage Area Heat Removal System

Heat is removed by the constant circulation of the basin water in direct contact with the waste canisters; heat is transferred to the secondary cooling loop and rejected to the atmosphere from standard open forced-draft cooling towers located just outside the storage building. A diagram of the system is shown in Figure 21-13.

FIGURE 21-13. WBC Heat Removal System

Each basin is equipped with a pump and heat exchanger system to maintain the basin water temperature at its operating level. Pumps and heat exchangers in the primary cooling loop are housed within the basin confinement structure with the heat exchanger and pump mounted as a unit on a support frame that can be quickly replaced by means of the building crane. Capability is provided to route the primary cooling loop through a series of filters and ion-exchange columns for particulate, chloride, or radionuclide removal on a routine basis. Similar facilities with larger capacity equipment are also provided for cleanup of a grossly contaminated basin.

Each basin and cooling system is isolated from the others to prevent cross-contamination in the event of canister or component failure within a storage system. An emergency water addition system provides water for shielding and cooling under all credible occurrences.

The temperature of the water in the basins is maintained at 49 °C. The temperature of the canister wall during storage in the water basin is 53 °C. The canister is fabricated of stainless steel,

Type 300 series and no general corrosion problem is anticipated for the canister in the water environment. Temperature and chloride content are two of the most important factors for causing stress corrosion cracking of stainless steels in a water environment, when uncontrolled. Therefore, the basin water temperature is controlled at a low temperature, 49 °C, and the chloride content of the water is maintained at less than 10 parts per million.

Storage Area Ventilation

Supply air to each storage basin is prefiltered to minimize dust and algae collection in the basin water, and is preconditioned, as necessary, for personnel comfort in the area. The air normally discharges directly to the atmosphere. However, in an emergency, the exhaust flow will be automatically diverted through a high-efficiency particulate air (HEPA) filter before discharge to the atmosphere via the stack.

A portion of the exhaust air is withdrawn from immediately above the pool area to prevent the accumulation of hydrogen from the radiolysis of water. The basin covers maintain a ventilation barrier between the building deck area and the water pools to minimize the heat load in the personnel work area.

Retrievability

The canisters are stored in double tiered racks in the storage basins. The racks are constructed to permit retrievability of all canisters with a minimum amount of extra movement of canisters. The canisters can be transferred back to the receiving cell, for either inspection or for shipment, by using the transfer cask. Overpackaging the canister can be accomplished in the receiving cell, if required.

Monitoring and Surveillance

A monitoring system is provided for taking routine measurements for the purpose of documenting compliance with existing regulations. The monitoring system consists of fixed monitors, recording continuously, and mobile monitoring performed by radiation protection personnel who take wipes and air samples at specified locations on a programmed basis.

A surveillance system is provided for containment verification. This is primarily achieved through observance of the facilities in terms of anticipating problems and detecting a possible release within the building. Radiation detection equipment is installed in the basin water recirculation line for each basin so that leaking canisters may be detected early and removed for repackaging.

21.2.2.2 Status of the Technology

The WBC is based on technology which is well known in the nuclear industry. The industry has had about 30 years of experience with water basins in spent fuel element storage service. The principal technical features are heat transfer, water corrosion, and radiation shielding design. No difficulties are anticipated in the application of these technologies to the particular needs of the WBC.

- Estimated Time of Readiness for Practical Design

Conceptual design engineering studies have been performed for the WBC alternative by ARHCO for ERDA in the Retrievable Surface Storage Program. About six to nine months would be required to complete conceptual design engineering. An additional nine to twelve months would be required to complete the design criteria, select an architect-engineer for preliminary engineering design, and start preliminary engineering design. With the proper approvals, design and construction could be completed so that facility startup could commence within six years after start of preliminary engineering.

- Major Development Program Objectives To Be Met

No major development work is required for the implementation of the WBC alternative. Sufficient experience has been accumulated in the nuclear industry over the years with water basins, where they have been used for reactor fuel elements storage, to obviate the need for development and permit implementation of the WBC for high-level waste storage service with a high degree of assurance.

21.2.2.3 Variant Alternative - Water Basin Concept With Overpack

A variance in the water basin concept (WBC) is the Water Basin Concept with Overpack (WBC-OP). The WBC-OP is similar to the WBC described above except that in the WBC-OP all incoming high-level waste canisters would be routinely overpacked in the receiving and assembly facilities prior to storage in a water basin.

A description of the WBC-OP conceptual design is presented below.

- Technical Description

The interim storage facility using the WBC-OP alternative for cooling the waste would consist of two principal areas. These are the receiving and assembly building and the storage area. In the receiving and assembly building waste canisters are received, encapsulated into storage units, and the storage units are placed into a transfer cask for transfer to the storage area. In the storage area the transfer

cask is received and unloaded under water. The canisters are stored under water in basins and are kept under surveillance for an interim period.

General

Operations in the receiving and assembly building are similar to those for the WBC, except that facilities for receiving, welding and testing of overpacks are needed for the WBC-OP. At the weld and test station each canister is encapsulated in a 12.5 mm-thick, stainless steel-type 304-L overpack to form a storage unit (see Figure 21-14). The storage units are moved to the storage area in a heavily shielded transfer cask and stored under water in a water basin similar to the WBC. The wall thickness of the overpack, 12.5 mm, is determined by the need to provide a rugged overpack which can withstand all of the required handling operations. General corrosion of the overpack is not expected to be a problem as the stainless steel is quite resistant to a water environment.

FIGURE 21-14. Water Basin Concept - Overpack, Storage Unit

Receiving and Assembly Building

The receiving and assembly building of the WBC-OP is basically the same as that for the WBC, with the exception that facilities to receive, weld and test the overpacks are needed. An overpack receiving and assembly area, and weld and test stations are provided.

Storage Area

The storage basins for the WBC-OP are similar to those provided for the WBC. The overpack provides an additional confinement barrier when comparing the WBC-OP to the WBC. Overpacking eliminates requirements for concrete cover blocks and eliminates the need for a two-loop water cooling system (see Figure 21-15) due to the low possibility of contaminating basin water by a failed storage unit. Also eliminated are the diversion features of the ventilation system in the storage area which provide capability to route the exhaust air through an HEPA filter prior to discharge to the stack. The retrievability features of the WBC-OP and the monitoring and surveillance systems provided are similar to those provided for the WBC (see Section 21.2.2.1).

FIGURE 21-15. Water Basin Concept - Overpack, Heat Removal System

- Status of the Technology

The status of the technology for the WBC-OP is similar to that discussed for the WBC (see Section 21.2.2.2) and for the Air-Cooled Vault Concept (ACVC) (see Section 21.2.3.2).

21.2.3 Air-Cooled Vault Concept (ACVC) Alternative

INTERIM STORAGE OF SOLIDIFIED HIGH-LEVEL WASTE AIR-COOLED VAULT CONCEPT

FLOW DIAGRAM

Storage Unit Description

Canisters of waste are overpacked in carbon steel and a cast iron shield plug is installed in the top of the overpack. The overpack is 406 mm OD x 381 mm ID x 4.3 m long, and is closed by welding a cap to the top of the overpack.

Applicable Wastes

Solidified High-Level Waste.

Process Description

Shipping casks containing canisters of waste are unloaded in a shielded facility, which is remotely operated. Canisters are examined and decontaminated, if necessary. In a partially shielded facility the canisters are overpacked and the overpack is closed by semiremote welding. Nondestructive examination of the overpack closure is performed, semiremotely. The storage unit is moved to the storage vault in a bottom loading transporter and placed into storage. The storage units are cooled by natural convection air flow through the vault.

Principal Advantages

No mechanical systems required to maintain cooling (passive system); canister is protected by the overpack and overpack corrosion will be minimal; coolant is readily available.

Principal Disadvantages

Canisters will have a relatively high internal temperature; loss of coolant flow will lead to rapid temperature increases; gross container failure could lead to rapid spread of contamination; and limited in ability to handle higher heat fluxes.

Development Status

No developmental work is required, but a demonstration program would be desirable to verify design calculations.

Process Effluents

Solid nuclear waste is placed into drums for disposal; radioactivity in the gaseous effluents is negligible, water vapor is released from the cooling

tower and from the evaporation pond; liquid effluents are collected in and evaporated from the pond; accumulated salts from the evaporation pond are trucked to a disposal area.

The Federal repository for interim storage of the solidified high-level waste would provide the facilities for safely processing and storing the high-level radioactive waste generated by the nuclear power industry for an interim period. Based on the results of recently completed studies, the ACVC is another method that could be applied to this service.

A description of the ACVC conceptual design follows.

21.2.3.1 Technical Description

The interim storage facility using the ACVC alternative for cooling the waste would consist of two principal areas. These are the receiving and assembly building and the storage area. In the receiving and assembly building waste canisters are received and placed in storage units. In the storage area the storage units are stored below ground in a concrete vault and are kept under surveillance for an interim period.

- ACVC Storage Unit

The ACVC storage unit consists of a high-level waste canister (see Figure 21-1), a carbon steel overpack and a cast iron shield plug installed in the top of the overpack (see Figure 21-16). The carbon

FIGURE 21-16. Air-Cooled Vault Concept - Storage Unit in Cell

steel overpack is 406 mm OD by 381 mm ID by 4.3 m long. The canister is placed inside of the empty overpack and a cast iron shield plug is installed in the top part of the overpack, about 1 meter above the top of the canister. The overpack is closed by welding a cap to the top of the overpack.

During storage in the vault the temperature of the carbon steel overpack is about 204 °C. At this temperature, general corrosion of the carbon steel in an air environment is estimated at about .05 mm per year. The storage unit is designed for a 100-year life, hence, the total corrosion allowance is 5 mm. The wall thickness of the overpack is 12.5 mm which provides sufficient ruggedness to withstand all of the handling operations required. In addition, retrievability of the storage units, which is discussed later, is provided so that overpacks can be replaced or the overpacks can be overpacked themselves in the weld and test cell.

- General (See Figure 21-17)

Shipping casks are received and the waste canisters are unloaded in the receiving area. Overpacks are prepared in the assembly area, then loaded into the transfer cask and moved to a position below the

FIGURE 21-17. ACVC Conceptual Layout

receiving cell. A canister and a cast iron plug are loaded into the overpack and the transfer cask is moved to the weld and test station. An overpack lid is then welded on the overpack, the weld is tested, and the transfer cask is moved to the loadout station.

A bottom-loaded shielded transporter raises the storage unit into its shield cavity. This is done through a port in the top of the loadout station. The transporter carries the storage unit to the storage area, which consists of a series of concrete vaults. The storage units are stored in vertical sleeves in the vault. The transporter positions itself over a sleeve, removes a concrete shield plug from the top of the sleeve, and lowers the storage unit into the sleeve.

Heated air in the annulus between the sleeve and the storage unit rises vertically through the vault and is discharged to the atmosphere via a raised exhaust port. The buoyant, vertical movement of hot air creates a negative pressure, drawing ambient air through the air intakes at the sides of the vault.

- Receiving and Assembly Building

The receiving and assembly building consists of the following areas: receiving area, weld and test area, overpack assembly area, and a loadout station.

Receiving Area

The receiving area is similar to that described in Section 21.2.1.1 for the SSCC alternative.

Weld and Test Area

The weld and test area consists of weld and test stations where the overpack containing a canister is closed by welding and the weld is tested for integrity. Operations at the weld and test stations are performed semiremotely and contact maintenance is performed on all of the equipment. Each station is equipped with a remotely operated welding head, a nondestructive test head and a repair head. Neutron shielding is provided for personnel performing the welding and testing work.

Overpack Assembly Area

The overpack assembly area is a high bay extension of the shipping cask receiving area. The overpack assembly area contains working platforms which serve as weld preparation and overpack inspection stations. Space is also provided for storage of empty overpacks.

Loadout Station

A drive-through loadout station is provided to allow the shielded transporter, which has limited maneuverability, to enter the building, be stationed directly above the storage unit loadout port, receive a storage unit, and drive directly from the building.

● Storage Area

The storage area consists of a concrete structure containing a fixed number of storage cells (vaults) which hold about 500 storage units each. Storage units are suspended from the roof of a cell to form a single-level array. The top of the deck is 1.2 meters above the surrounding grade to minimize ramps for the transporter and yet protect the air intakes from wind blown debris. (See Figure 21-18.)

FIGURE 21-18. ACVC Vault Section

Concentric metal pipe sleeves about 508 mm in diameter are provided for each storage position. The sleeves are attached to a tube sheet located 0.9 meters above the floor of the vault to create a plenum to equally distribute the incoming air to the sleeves. Means are provided to plug off sleeves which are not in use. Tube sheets,

sleeves, and structural supports in the vault are fabricated of unpainted low carbon steel.

Periodic inspection and cleaning will be required to prevent a buildup of dust and dirt in the bottom of the cells. Protective screens and covers for the air inlet and exhaust ports are provided to prevent the entry of large foreign objects.

Transporter

The transporter is used to move a storage unit from the loadout station in the receiving and assembly building to the storage area. Personnel protection against radiation from the storage unit is provided on the transporter by the use of heavy shielding. It is equipped with cooling capability to carry one storage unit up to 24 hours. The transporter is designed for remote placement of a storage unit with supplemental viewing provided to assist in removing the shield plug from the storage vault, placing the storage unit into the vault, and replacing the shield plug. It is also capable of performing the reverse of these operations to retrieve storage units.

Retrievability

The storage units in the vault can be retrieved through use of the transporter and returned to the receiving and assembly building. The storage units would be received by the transfer cask and transported to either the weld and test area or to the receiving cell. Capability to retrieve the waste canister from the overpack, to re-overpack the canister and to prepare the canister or the overpack for shipment is provided.

Monitoring and Surveillance

A monitoring system is provided for taking routine measurements for the purpose of documenting compliance with existing regulations. The monitoring system consists of fixed monitors, recording continuously, and mobile monitoring performed by radiation protection personnel who take wipes and air samples at a specified location on a programmed basis.

A surveillance system is provided for containment verification. This is primarily achieved through observance of the facilities in terms of anticipating problems and detecting a possible release at the storage vault level in time to contain the release within the confines of the site. The storage area on-line surveillance system provides immediate detection of a radioactive release, positive identification of failed storage units, and possible detection of impending storage unit

failures, which would be accomplished with instrumentation other than radioactivity detectors and/or with a nondestructive examination program.

21.2.3.2 Status of the Technology

The ACVC is based on technology which is well known in the nuclear industry. Heat dissipation by the application of air-cooling has been used by the nuclear industry in many different ways over the years. The principal technical features are heat transfer, corrosion in an air environment, fluid flow, and radiation shielding design. No difficulties are anticipated in the application of these technologies to the particular needs of the ACVC.

- Estimated Time of Readiness for Practical Design

Conceptual design engineering studies have been performed for the ACVC alternative by ARHCO for ERDA in the Retrievable Surface Storage Program. About six to nine months would be required to complete conceptual design engineering. An additional nine to twelve months would be required to complete the design criteria, select an architect-engineer for preliminary engineering design, and start preliminary engineering design. With the proper approvals, design and construction could be completed so that facility startup could commence within six years after start of preliminary engineering.

- Major Development Program Objectives To Be Met

No major development work is required for the implementation of the ACVC alternative. However, it may be desirable to demonstrate the concept feasibility and the safety features. The primary objectives of a proposed demonstration program would be as follows:

- Fabricate storage units and construct a miniature air-cooled vault. Operate the air-cooled vault under nonradioactive conditions for design verification.
- Demonstrate the structural integrity of the transfer cask, and the storage unit sleeves in the vault.
- Construct and monitor an air-cooled vault containing canisters of high-level waste to obtain operating experience.
- Demonstrate the storage unit surveillance instrumentation systems.

21.2.3.3 Variant Alternative - Forced Air-Cooling Concept (FACC)

A variance in the Air-Cooled Vault Concept (ACVC) is the Forced Air-Cooling Concept (FACC). The FACC is similar to the ACVC described above except that HEPA filters would be provided to filter all of the exhaust air from the vault and fans would be provided to motivate the air through the vault and filters and out the stack.

- Technical Description

The interim storage facility using the FACC alternative for cooling the waste would consist of two principal operating areas. These are the receiving building and the storage area. In the receiving building waste canisters are received and are transferred to the loadout station for transport to the storage area. In the storage area the canisters are stored in heavily shielded cells, thick concrete vaults, and are kept under surveillance for an interim period.

- General

The receiving building of the FACC is basically the same as that for the WBC (see Section 21.2.2.1), with the exception that a loadout station is provided for the FACC for pickup of the canister by the transporter.

- Storage Area

Canisters are stored in the concrete vault in a manner similar to that discussed for the ACVC. The storage units for the FACC are bare canisters instead of overpacked canisters as are used in the ACVC. In place of the barrier furnished by overpacking, HEPA filters are provided to filter all exhaust air from the vault. Fans are installed to motivate the air past the canisters, through the HEPA filters, and out of the stack. The transporter, the retrievability features, and the monitoring and surveillance system provided for the FACC are all similar to those provided for the ACVC (see Section 21.2.3.1).

- Status of the Technology

The status of the technology for the FACC is similar to that discussed above for the ACVC (see Section 21.2.3.2).

21.2.4 Variant Alternative - Vault Storage with No Air Flow - For Aged Wastes

The air-cooled concepts previously discussed are based upon the premise that the age of the solidified high-level waste to be stored is about ten years after reprocessing. Depending upon the heat density of the canister, which can be varied by adding inerts, or intermediate-level waste to the solidified high-level waste in the canister, there is a time after ten years when the waste heat has decayed to such a low level that it is not necessary to have air flowing past the storage unit for safe heat removal at all times. Of course, this could be designed to occur at time periods of ten years or less, but the penalty for such conditions is the addition of sufficient inerts to dilute the high-level waste to such a low level, and hence, increase the volume of total waste to be packaged and stored as high-level waste.

A cooling concept which requires a minimum amount of operator attention, because no air flow is used, can be designed so that the heat from the storage unit would be dissipated by radiation, conduction, and convection to its surroundings in a vault. The heat density of the waste would determine the geometric layout of the storage units in the vault. Certainly, any waste can be safely stored in such a manner by spreading out the storage units sufficiently in a geometric pattern such that there is no or little amount of heat interaction between individual storage units. For ten-year-old high-level wastes packed to a reasonable heat density this is uneconomical. The spread is so great that the amount of vault space required is prohibitive in economic terms.

Engineering studies need to be conducted to study the optimum heat densities and geometric spacings for the various wastes. Sufficient work has been done in the past to indicate that viable alternatives can be developed by permitting the waste to age, for instance in the FRP water basin, long enough to allow cooling by simpler methods than those previously discussed (see Sections 21.2 through 21.4). The principal air-cooled concepts previously discussed require constant operator attention. In the SSCC, attention is needed to assure that the inlet and outlet cooling ports are not plugged with dirt, sand, or tumbleweeds, for extended periods of time. In the ACVC it is also important to keep the air inlet ports free from debris. In the FACC operator attention is needed to assure that the fans are always operating.

The heat transfer technology involved is well understood in the nuclear industry. The application of the technology needs to be performed based on specific wastes, specific heat densities, and specific sizes of canisters.

21.3 UNDERGROUND FACILITY (NEAR SURFACE) FOR INTERIM STORAGE OF SOLIDIFIED HIGH-LEVEL WASTE ALTERNATIVE

An underground facility (near surface) for interim storage of solidified high-level waste would provide additional isolation of high-level waste from the general public, and additional physical protection of the waste canisters by virtue of its underground location. It would be more resistant than the previously discussed concepts to attempted sabotage or terrorist action at ground level due to the separating layer of earth material. The entrances to surface facilities and shafts leading to subsurface facilities would be protected from entry of potential saboteurs by physical security systems and administrative controls similar to those intended for a surface facility. The subsurface facilities would be resistant not only to sabotage, but also to the impact of falling objects (e.g., airplane crashes, meteorites). The subsurface storage concept would also provide additional radiation shielding for persons at ground level due to the earth cover.

21.3.1 Cooling Concept for Underground Facility

The FACC is recommended for the underground facility for interim storage of the solidified high-level waste. Its application provides the most complete underground facility. The fans which provide the motive power to the cooling air stream can easily be operated underground. The two sealed storage cask concepts (SSCC and SSCC-MW) are basically surface facilities which depend upon the open atmosphere for dissipation of heat and radiation and as such are not amenable to being enclosed in a structure. The two water basin concepts (WBC and WBC-OP) depend upon water cooling towers for dissipation of the heat to the atmosphere and these should be located at the surface. The ACVC would need a tall stack to induce air flow for cooling and it would be located at the surface.

21.3.2 Technical Description

The receiving area, where the shipping casks containing high-level waste canisters are received, is located at the surface. The shipping cask unloading area, where the shipping cask is opened and unloaded, is underground. The storage vault area where the canisters are stored and all essential support services are located underground. Access controls to the surface facilities are provided by normal means such as security guards and fencing.

21.3.2.1 Shipping Cask Receiving Area

Shipping casks containing high-level waste canisters are received in the shipping cask receiving area, which is constructed at ground level. The shipping cask is removed from the rail car or truck, decontaminated, if required, and lowered down a shaft into a transfer gallery in the shipping cask unloading area.

21.3.2.2 Shipping Cask Unloading Area

The shipping cask unloading area is located below ground level and located within this area are the transfer gallery and the transfer cell. The shipping cask is transported within the transfer gallery to a position beneath the transfer cell. It is then elevated to seal to the bottom of the transfer cell. After removal of the shielding and sealing plugs in the transfer cell door and in the top of the shipping cask, the waste canisters are lifted individually into the cell, inspected, checked for contamination and decontaminated, if required. All operations within the transfer cell are conducted remotely by use of the transfer cell crane and manipulators. Canisters are then moved individually into the storage area canister receiving station.

21.3.2.3 Storage Area

The storage area is located below ground level and consists of a series of vaults, each of which can store 3,000-5,000 canisters. The canisters are

transferred from the storage area canister receiving station into a bottom-loading shielded transporter. The transporter is driven through a transport corridor to the appropriate storage vault.

Canisters are stored in the concrete vault in a manner similar to that described for the ACVC. The storage units for the underground storage concept are bare canisters instead of overpacked canisters as are used in the ACVC. Fans are installed to motivate the air past the canisters, (to remove canister heat) through HEPA filters, and out into the atmosphere. The transporter, the retrievability features, and the monitoring and surveillance system provided for the underground storage concepts are all similar to those provided for the ACVC (see Section 21.2.3.1).

21.3.3 Status of the Technology

The status of the technology for designing and operating the underground storage concept is the same as discussed for the ACVC (see Section 21.2.3.2), except for the features of performing underground construction work. Conventional mining and conventional underground construction techniques would be utilized. No unusual technical problems are foreseen in the application to the construction of an underground storage facility. Program timing and development needs are also the same as discussed for the ACVC (see Section 21.1.3.2).

21.4 ALTERNATIVE LOCATIONS FOR INTERIM STORAGE OF SOLIDIFIED HIGH-LEVEL WASTE

The FRP will package the solidified high-level waste into canisters and then store them at the FRP until they are shipped to a Federal repository, where they are then placed into Federal custody. The present Federal regulations, 10 CFR 50, Appendix F, require the FRP to convert liquid high-level wastes to solids within five years after reprocessing and to ship the solidified waste to a Federal repository within ten years after reprocessing.

Currently there are no Federal repositories as no final geologic storage sites have been designated to accept such waste. Such sites may not be designated by the time that the FRPs will be ready to ship the first canisters of solidified high-level waste. Hence, interim storage of the solidified high-level waste may be necessary during the time that extends from when the FRPs are ready to ship the waste and when final geologic storage is ready to receive the waste.

Interim storage can be accomplished at the FRP, at a single national site, at regional sites, or at final geologic storage sites. Studies need to be conducted on all these alternative locations to determine the economic impact of each (see Figure 21-19).

FIGURE 21-19. Location Alternatives for Interim Storage of HLW

21.4.1 Interim Storage at FRP

Interim storage could be accomplished at the FRP and either the FRP or ERDA could have custody of the waste. The intent of these alternatives is to minimize the number of transports of waste in the public domain. The total number of transport miles for the waste would be a minimum. There would be as many interim storage facilities as there would be FRP sites dispersed throughout the country. Even though more than one FRP may be located at the same site, ultimately only one interim storage facility would be needed to serve them.

21.4.1.1 Interim Storage at FRP With Custody of the Waste by FRP Alternative

As mentioned previously, current regulations require the FRP to ship the solidified high-level waste to a Federal repository within ten years after reprocessing. An alternative to this is for the FRP to retain custody of the waste for longer than ten years. Depending upon the time that the FRP retains custody, transfer of the waste would be to either final geologic storage or to

an interim storage facility where only a simple, passive, no air flow system is required for cooling the waste, as discussed in Section 21.2.4.

If the FRP retains custody of the waste until final geologic storage is ready to receive the waste, the storage time required could be in tens of years. One of the cooling concepts discussed above could be used for interim storage, or a combination of them could be used. This would require additional facilities and space that the FRP would need to furnish. The FRP would be responsible for designing and obtaining the necessary licensing required for such facilities.

If the FRP retains custody of the waste only long enough to satisfy the conditions discussed in Section 21.2.4, the additional facilities required most likely would only be an expansion of its water basin for extra storage capacity. This variant alternative would be of lesser impact on the FRP than interim storage for tens of years.

21.4.1.2 Interim Storage at FRP With Custody of the Waste by ERDA Alternative

Interim storage at the FRP can be provided by ERDA by locating a Federal repository at the FRP. The intent of this alternative is for ERDA to assume and retain custody of the waste until final geologic storage is ready. The FRP would provide space to ERDA on its site for the facilities. ERDA would be responsible for the design and the licensing requirements of the facility.

This alternative would be the most expensive approach for ERDA in considering alternative interim storage locations. As each FRP site would house a Federal repository, the number of repositories would be a maximum. Capital and operating costs would be a maximum compared to the other alternatives. The obvious advantage to be gained is that only one transport of high-level waste would be made offsite to final geological storage.

21.4.2 Interim Storage at Regional Sites Alternative

Interim storage can be provided on a regional basis so that the FRPs in a region all ship their waste to one regional Federal repository. Such a repository could be located at an FRP site, or a Federal site. Transportation for the near-term would be less to regional sites than to a single national site. The use of regional interim storage sites could also develop into the use of regional final geologic storage sites.

The alternative of regional sites is a more expensive approach than the alternative of a single national site, both in capital cost and in operating costs.

21.4.3 Interim Storage at a Single National Site Alternative

The alternative for having one single Federal repository for interim storage results in the minimum capital and operating cost expenditure for

interim storage. Waste from all the FRPs throughout the country would be sent to the Federal repository. When a final geologic storage site has been selected for waste disposition, the interim storage repository would be emptied and all of the waste moved to final storage. It is also possible that the final geologic storage site could be located at the same site where the interim storage Federal repository is located. This would certainly minimize the impact upon transportation of the waste between the two Federal repositories.

21.4.4 Interim Storage at the Final Geologic Storage Site(s) Alternative

The alternative of locating interim storage at the final geologic storage site would eliminate one transport of the high-level waste across the public domain. Should more than one final geologic storage site be selected, then there would be as many interim storage facilities as final storage sites. This approach would be more expensive than that of the single national site for interim storage, but it would be less expensive than the other alternatives (interim storage at the FRPs or at regional sites).

Timing would be an essential feature of this alternative. As previously mentioned, final geologic storage may not be ready for tens of years. If this alternative were followed, it would be on the basis of tentatively selecting a final geologic site and withholding final approval until sufficient development and demonstration activities had been completed satisfactorily.

Two tables compare the various alternatives for the interim storage of solidified high-level waste. Table 21-1 compares the facility requirements and Table 21-2 compares waste effluents and resource consumption.

TABLE 21-1. Comparison of Facility Requirements for Solidified HLW Interim Storage Alternatives(a,b)

<u>Cooling Concept</u>	<u>Weld and Test(c) Stations</u>	<u>Land Use, Minimum</u>	<u>Acres(d) Maximum</u>
Sealed Storage Cask Concept	1-5	100	1100
Water Basin Concept	None	25	100
Air-Cooled Vault Concept	1-3	25	100
Sealed Storage Cask Concept With Medium Wall	3-9	770	2300
Water Basin Concept With Overpack	1-3	30	150
Forced Air Cooling Concept	None	25	100
Vault Storage With No Air Flow For Aged Wastes	1-3	25	100
Underground Facility With FACC (Near Surface)	None	25	100

NOTES:

- a. Assumptions: Interim storage facility is sized to handle solidified high-level waste from 1500 MTU/year reprocessing plant, and store the waste for 10 years. (5 MTU/day with 82 percent time on-line). Solidified HLW has composition of reference waste (Section 6.0). Receipt rate and processing rate of canisters is according to the throughput considerations listed in Section 6.0.
- b. The receiving capacity of the interim storage facility is greater than the amount of waste that would be received from one FRP, by virtue of its basic design. For all of the cooling concepts listed above, except for the WBC and the WBC-OP, the receiving capacity is the total waste from about nine FRP's, while for the exceptions it is about five FRP's. The amount of waste from each FRP is based on Note a.
- c. A weld and test station consists of facilities and equipment for performing welding, stress relieving, and nondestructive examination work on storage casks or overpacks.
- d. The minimum land requirements are for a minimum size interim storage facility with waste storage capacity for about ten years production of one FRP; the maximum land requirements are for an estimated maximum size interim storage facility, or a minimum size interim storage facility with waste storage capacity for about 100 years production of one FRP.

TABLE 21-2. Waste Effluents and Resource Consumption for Solidified HLW Interim Storage Alternatives(a)

Cooling Concept	Waste Effluents(b)						Stainless Steel Steel (Tonnes)
	Radio- Activity In Gases	Liquids To Evap. Ponds (Ml/yr)	Non- Radioactive Solids (m ³ /yr)	Radio- Active Solids (m ³ /yr)	Elect. (kw-yr)	Water (Ml/yr)	
Sealed Storage Cask Concept	Negligible	60	240	60	1000	115	12
Water Basin Concept	Negligible	115	360	90	2000	230	130
Air-Cooled Vault Concept	Negligible	60	240	60	1000	115	12
Sealed Storage Cask Concept With Medium Wall	Negligible	60	240	60	1000	115	12
Water Basin Concept With Overpack	Negligible	115	300	75	1500	230	250
Forced Air Cooling Concept	Negligible	60	240	60	1400	115	12
Vault Storage With No Air Flow For Aged Waste	Negligible	60	240	60	1000	115	12
Underground Facility With FACC	Negligible	60	240	60	1500	115	12

NOTES:

- a. Assumptions: Interim storage facility is sized to handle solidified high-level waste from 1500 MTU/year reprocessing plant, and store the waste for 10 years.
(5 MTU/day with 82 percent time on-line).
Solidified HLW has composition of reference waste (Section 6.0). Receipt rate and processing rate of canisters is according to the throughput considerations listed in Section 6.0.
- b. Ml/yr = million liters per year.
m³/yr = cubic meters per year.

REFERENCES

1. Code of Federal Regulations, Title 10, Part 50, Appendix F (10 CFR 50, Appendix F).
2. Atlantic Richfield Hanford Company and Kaiser Engineers, Retrievable Surface Storage Facility Alternative Concepts, Engineering Studies, ARH-2888 REV, July 1974.
3. Panel on Engineered Storage, Committee on Radioactive Waste Management, National Research Council, National Academy of Sciences, Washington, D.C., Interim Storage of Solidified High-Level Radioactive Wastes, 1975.

22.0 TRANSPORTATION ALTERNATIVES

22.0 TRANSPORTATION ALTERNATIVES

22.1 INTRODUCTION

This chapter describes alternatives for the transportation of waste and fuel materials associated with the back end of the nuclear fuel cycle. Transportation methods will be described for:

- high-level waste
- spent fuel
- plutonium
- cladding hulls
- non-high-level waste.

These materials are produced in reactor operations, reprocessing of spent fuel, fabrication of fuel elements, and treatment and storage of radioactive wastes.

The operating and planned facilities to accomplish the fuel cycle objectives are currently situated in diverse geographic locations. Radioactive materials in a wide variety of physical and chemical forms must be transported between these facilities.

Many types of radioactive material are shipped routinely today. As the nuclear industry expands, the amount of these materials requiring transportation will increase. Transportation may be required in the future for other radioactive materials or for altered forms of materials presently being shipped. This chapter describes methods presently available or under development to meet current and near-term requirements for the transportation of radioactive material associated with the back end of the nuclear fuel cycle.

Figure 22.1. illustrates the role of transportation in the back end of the nuclear fuel cycle. Types of waste/fuel material requiring transportation are indicated in the figure along with their points of origin and destination. Treatment options which could have an effect on transportation requirements are also shown. In some cases, these treatment options will affect transportation methods while on other cases they affect only the transportation destination.

Transportation is shown for five basic waste/fuel materials:

- High-Level Waste. High-level waste is the primary waste generated when spent fuel is reprocessed. It contains nearly all of the fission products, 0.5% of the uranium and plutonium, and the majority of the other transuranics originally present in the irradiated fuel. High-level waste is generated as a liquid and solidified onsite after a maximum storage period of 5 years. Within 10 years from the time the fuel is processed,

FIGURE 22.1. Transportation in the Back End of the Nuclear Fuel Cycle
(Dashed Lines Indicate Possible Options)

solidified high-level waste must be shipped from the reprocessing plant site to a Federal high-level waste facility. It may be shipped directly to a high-level waste disposal facility or it may be shipped to an interim storage facility and subsequently transferred to a disposal site.

- Spent Fuel. Irradiated nuclear fuel is usually stored at the reactor for at least 120 days after reactor discharge. It is then shipped from the reactor to a chemical separations plant for reprocessing. There are proposals to establish separate fuel storage basins for spent fuel. Depending on the location of these facilities, an extra transportation step could be required. If spent fuel is not reprocessed, it would be shipped to a long-term storage facility.
- Plutonium. Plutonium is separated from spent fuel at the reprocessing plant. It is separated as plutonium nitrate solution, but (after 1978) it must be converted to a solid, expected to be plutonium dioxide, prior to shipment. It may be shipped to a fuel fabrication facility for the manufacture of mixed oxide fuel elements or to a storage facility. Subsequent shipments from the storage facility to a fuel fabrication facility will also be required.

- Cladding Hulls. Cladding hulls are removed from the spent fuel at the reprocessing plant. A proposed rule requires shipment within 5 years. They are expected to be stored onsite for a period of 1-10 years, then transported to a disposal site or to a facility for treatment and possible recycle of the Zircaloy.
- Non-High-Level Wastes. Non-high-level wastes are generated at reactors, chemical separations plants, fuel fabrication plants and at waste treatment facilities. These wastes are in a variety of forms including air filter media, water filtration resins and sludges, protective clothing and other combustible and noncombustible solids. Under a proposed rule, wastes contaminated with transuranic (TRU) elements will be shipped to a Federal storage facility. Other wastes are transported to a commercial burial ground. Fission product noble gases and tritium may be collected at the chemical separations plant and transported to a storage facility.

The principal transport modes for waste/fuel materials from the back end of the nuclear fuel cycle are truck and rail. In special cases, shipments by barge have been made. Barge transportation has some potential advantages that will stimulate investigation of its use for transporting fuel cycle materials in the future.⁽¹⁾ Presently air shipments of radioactive material are restricted to small quantities of radiopharmaceuticals and to radioisotopes used for research. The use of lighter-than-air ships has been proposed for future transportation of radioactive materials.⁽²⁾ Pipeline shipments for short distances have also been considered.

The majority of radioactive material shipments are presently carried on trucks. They provide a flexible and economic mode for the transport of a wide variety of materials. Truck shipments, however, are limited by state governments to a maximum payload weight of about 22 tonnes. Shipments exceeding this weight must be made in accordance with overweight permits issued by state agencies. Payloads of 100 tonnes or more can be accommodated in rail shipments. In many cases, improved payload to container weight ratios can be achieved in rail shipments because of the increase in size and weight of the shipping containers which can be transported by rail.

Shipments of radioactive material must be made in compliance with Federal, state, and local regulations. Federal transportation regulations establish container specifications, radiation dose rate limits and handling procedures that insure the safety of the public and transportation workers during the shipment of radioactive materials. State agencies regulate vehicle sizes and weights and in some cases transportation routes and times of travel. Federal regulations pertaining to the shipment of radioactive materials are summarized in Appendix E. The physical properties of some fuel cycle materials can pose unique constraints on the design of shipping containers and transport modes that comply with those transportation regulations. Spent fuel, for example,

requires containers with large amounts of shielding and a large heat removal capacity. Shipment quantities must be limited to keep container weights and dimensions within allowable limits for truck or rail transport. Shipments of plutonium must be made in special vehicles or under special transport conditions designed to prevent the diversion of fissile radioactive material.

The sections that follow present the technical alternatives for transporting the radioactive materials associated with the back end of the nuclear fuel cycle. Available technology, i.e., transportation systems presently in use are discussed in some detail. Information is also presented on transportation systems under development, i.e., those for which engineering principles exist and for which some preliminary design and/or testing has been carried out. In at least two cases, transportation of solidified high level waste and transportation of cladding hulls, the development of commercially available technologies has been controlled by need rather than by engineering.

22.2 SUMMARY

Transportation forms a vital link in the LWR fuel cycle. Transportation requirements for radioactive materials in the fuel cycle will continue to grow as the fuel cycle is completed and the LWR industry expands. Spent fuel, non-high-level waste and plutonium with low heat generation rates can be transported with low risk in credible accident conditions utilizing currently available technology. Shipping containers for the remainder of the fuel cycle materials (high-level waste, cladding hulls and Pu requiring radiation shielding and larger heat removal capabilities) have not been developed because there has been no need to transport these materials. Conceptual design studies have been completed for packaging these materials. No physical reasons have been discovered that would prevent the design and construction of containers to safely and efficiently transport these materials when required. Basically these containers would represent an adaptation of currently available technology to meet the transportation requirements of a particular waste form.

The transportation of radioactive materials is regulated by the U.S. Department of Transportation and the Nuclear Regulatory Commission. Their regulations delineate packaging requirements, allowable radiation dose rates and handling procedures to insure that shipments of radioactive material are made in a manner that poses minimum risks to the public and transportation workers. The major effort in developing radioactive material shipping containers is the design of a container to safely, efficiently and economically transport material within the framework of these regulations.

Table 22.1 summarizes transportation requirements in the back end of the LWR fuel cycle. The amount of material requiring transportation is presented along with the total number of required shipments using representative

TABLE 22.1. Summary of Information on Waste/Fuel Material Shipments
(based on 1500 MT/year reprocessing plant)

WASTE/FUEL MATERIAL	ANNUAL (a) QUANTITY SHIPPED	PACKAGE (b)		NUMBER OF PACKAGES (h)	TRANSPORT VEHICLE			AVAILABLE TECHNOLOGY		TECHNOLOGY UNDER DEVELOPMENT	
		TYPE OF PACKAGE	QUANTITY PER PACKAGE		TYPE OF VEHICLE	QUANTITY PER SHIPMENT	NUMBER OF SHIPMENTS	PACKAGE	TRANSPORT VEHICLE	PACKAGE	TRANSPORT VEHICLE
High-Level Waste	90 m ³	Cask	2.4m ³	38	Train	1 Cask	38	No commercial packaging available	Rail transport similar to that used for spent fuel elements	Design studies of canister and casks for shipment of solidified waste	Current technology adequate
Spent Fuel	4800 Fuel Assemblies	Truck Cask	1 PWR or 2 BWR Assemb	3600	Truck	1 Cask	3600	Shipping casks in use for both truck and rail shipments	Truck shipments use specially designed semi- trailer. Rail shipments on 100 ton flat car	Design of casks to accom- modate large number of high burnup fuel elements	Current technology adequate
		Rail Cask	7 PWR or 18 BWR Assemb	476	Train	1 Cask	476				
Plutonium	13.2 MT of PuO ₂	Canister with Overpack	32 kg	412	Truck	7 Packages	59	Containers in use for commercial transport of plutonium from low burnup fuel	Truck transport in exclusive use vehicles	Design and testing of containers for large quantities of high exposure material	Truck with immobilization penetration resistance, and continuous communication capabilities
Cladding Hulls	487 MT	Cask	1620 kg (e)	300	Train	1 Cask	300	No containers specifically designed for cladding hulls Adaptation of existing containers possible	Truck or rail depending on size of cask	Design studies for casks to carry untreated hulls Conceptual design for compacted hulls cask	Current technology adequate
Non-High- Level Waste								Type of package dictated by form of waste. Packaging available for all waste forms	Shipment by common carrier	Current technology adequate	Current technology adequate
• Dry Solids											
• Combustible	15,208m ³	Metal drums	0.21 m ³ (c)	1.67x10 ⁵	Truck	64 Drums	2604				
• Non-Combustible	3.951m ³	Metal bins									
• Solidified Liq	15,844m ³ (d)	Wooden boxes									
• Filters		Shielded & Unshielded Overpacks			Train	140 Drums	1191				
• Noble Gases	84 Cylinders (g)	Gas Cyls in shielded casks	50 liters	84	Train	6 Casks	14	Noble gases are not currently transported except for research purposes. A krypton shipping container is available	Truck or rail depending on weight of shipment	Conceptual designs for rail cask	Current technology adequate
• Tritium	16 Sm ³ (f)	Drum	0.0045m ³	3700	Truck	64 Drums	58				

NOTES

- a) Extracted from Chapter 2. Untreated waste unless otherwise indicated.
b) Packages listed are either currently available or expected to be developed in the near future.
c) 55 gallon drum.
d) Assumes volume increase of 2 in solidification process.
e) Based on AGNS conceptual cask for clad from 5 MTU spent fuel.
f) Solidified as Ca(OH)₂.
g) Cylinder pressure 154 Kg/cm².
h) Assumes entire annual quantity shipped in containers indicated.

containers and transport vehicles. The table is scaled to the annual requirements for a 1500 MTHM*/year reprocessing plant. This plant provides the annual reprocessing capability for 22 Pressurized Water Reactors and 11 Boiling Water Reactors. A 1500 MTHM/year fuel fabrication capacity is also required for this unit fuel cycle. The waste quantities presented in the table represent untreated waste unless otherwise specified. The status of the technology of transportation modes and shipping containers presently available and under development is also indicated for each waste form. As indicated in the table, most shipments of radioactive material are made by rail or truck. This is expected to continue in the future.

Transportation of spent fuel, solidified high-level waste, plutonium, cladding hulls and non-high-level waste is discussed in this chapter. A brief summary is presented below for the technology presently available and under development for each waste form.

- Solidified High-Level Waste has not been transported to date. Transportation of solidified high-level waste is expected to be a future requirement of the nuclear industry. Conceptual designs exist for heavily shielded shipping casks to transport this waste form. These casks will be similar to currently available spent fuel casks. They are expected to be transported primarily by rail because of their weight. No technical obstacles have been identified to date that would prevent the design and construction of solidified high-level waste shipping containers when they are required.
- Spent Fuel has been shipped in the U.S. for many years. Massive heavily shielded shipping casks are available for both truck and rail transport of high burnup, short cooling time fuel from current generation LWR's. Currently available technology is adequate to accommodate future requirements of the Light Water Reactor industry.
- Plutonium is presently shipped routinely in small quantities. Most shipments are made by truck. Federal requirements to prevent the diversion of fissile material during transportation necessitate the use of secure transportation systems that may include armed escorts, continuous radio communications with the transport vehicle and vehicle immobilization features. Shipping containers are presently available for plutonium that require no shielding and have a relatively low heat generation rate. Containers are under development for Pu from higher burnup LWR fuels that will contain larger percentages of ^{238}Pu , requiring containers with neutron shielding and greater heat removal capabilities.

* Metric tons heavy metal (U+Pu).

- Cladding Hulls have been disposed of by onsite burial at the reprocessing plant. There are no shipping containers presently available that are specifically designed for the transportation of cladding hulls. It may be possible to ship cladding hulls by adaptation of some existing general purpose containers. Containers are under development for transportation of untreated hulls from high exposure LWR fuel with short cooling times. Several treatment processes have been proposed for cladding hulls. Transportation requirements are dependent on the particular process or combination of processes that might be chosen.
- Non-High-Level Waste is generated at all nuclear facilities. It can have a variety of physical forms and radiological properties. A variety of treatment processes are also possible for this waste. The technology to transport non-high-level waste exists today. Special radioactive material shipping containers may be required for certain waste forms, but they may be built by adapting existing technology. The majority of non-high-level waste has relatively low radiation levels and can be shipped in DOT specification containers. Most shipments will be carried by truck.

Transportation generates only minor quantities of wastes requiring treatment or disposal. The handling operations required to prepare radioactive material for shipment generate small amounts of waste, but these are mostly low level and can be treated with normal plant wastes. Packaging used to transport radioactive material is generally disposed of with the waste or reused. The increased number of shipments of radioactive material that will be required as the LWR industry expands will not of themselves add significantly to the amount of radioactive material requiring disposal.

22.3 TRANSPORTATION OF HIGH-LEVEL WASTE

22.3.1 Introduction

High-level waste is the byproduct of the first solvent extraction cycle and the concentrated wastes from subsequent extraction cycles when irradiated reactor fuels are chemically reprocessed. High-level waste contains essentially all of the nonvolatile fission products, about 0.5% of the uranium and plutonium, and all of the other transuranic actinides originally present in the spent fuel. The composition of high-level waste is shown in Table 2.8.

High-level waste will be solidified at the reprocessing plant before it is shipped to a Federal repository. Solidification processes are described in Section 6.1. After solidification the waste will either be shipped directly to a waste disposal facility or to an interim storage facility and subsequently transferred to a disposal site.

Several recently published documents^(3,4) provide information on policy and procedures for the transportation of high-level nuclear waste.

22.3.2 Considerations Important to Transportation

Regulatory Constraints

Current Federal regulations⁽⁵⁾ require that high-level liquid radioactive wastes be converted to a dry solid not later than 5 years following separation of the fission products from the irradiated fuel. The solidified waste must be shipped from the reprocessing plant to a Federal repository not later than 10 years from the date of separation of the waste from the fuel.

From a decay heat standpoint it would be advantageous to postpone shipment of the waste until near the end of the 10-year storage time currently allowed at the reprocessing plant. After 10 years storage time, the decay heat of the fission products will be only about 5% of the level present when the spent fuel was originally processed. (See Section 2.4.)

Shipments of solidified high-level waste must conform to Federal regulations for the shipment of radioactive material. These regulations are summarized in Appendix E.

Significant regulatory constraints which affect high-level waste shipments include:

- Type B packaging capable of satisfying the accident test criteria of 10 CFR 71 and 49 CFR 173.398.
- Limitation of the dose rate to 10 mrem/hr at a distance of 6 ft from the external surface of the transport vehicle.
- Limitation of the accessible cask surface temperature to 82°C.

Physical Constraints

The 10-year old waste from processing a tonne of reference LWR fuel will have a total activity of 2.51×10^5 Ci and a heat generation rate of 1.32 kW. To provide sufficient shielding for high-level waste shipments it will be necessary to utilize shipping casks weighing many tonnes.

Most high-level waste shipments will be made by rail. In order to remain within the legal weight restrictions of most states, a cask shipped by truck would have to be limited to a weight of about 22 MT. Casks for rail shipment will probably weigh about 100 MT with the weight limited by crane lifting capacities at reprocessing plant and repository sites. Advantages of the larger weight rail cask over the smaller truck cask include:

- Increased efficiency of shipment (measured in kilograms of payload per kilogram of total cask weight).

- Reduction in the number of handlings at the reprocessing plant and the Federal repository.
- Reduction of the total inventory of casks needed to ship high-level waste.

22.3.3 Available Packaging and Transportation System

High-level waste is not currently being commercially solidified. Casks designed specifically for shipment of solidified high-level waste have not been built.

High-level waste shipping containers are expected to resemble the casks currently available for rail shipment of spent fuel from a reactor to a reprocessing plant. Heat dissipation and protection from high radioactive dose rates which are major considerations in the shipment of high-level waste are also prime considerations in the shipment of spent fuel. Thus the shipment of high-level waste will benefit from the extensive experience gained in shipping spent fuel elements.

22.3.4 Packaging and Transportation Systems Under Development

Packaging

Current research (see Section 6.0) indicates that high-level waste will be solidified in cylindrical canisters at a high temperature and that the resultant solid will be thermally stable. The waste canisters will probably be constructed of 304L stainless steel. Other candidate materials are also being investigated. Probable design parameters for waste canisters will include:

- diameter, 16 to 60 cm
- length, 3 to 4.5 m
- maximum heat density, 2.9 kW/linear meter
- radiation limits, 1×10^3 rem/hr neutron dose rate and 1×10^6 rem/hr gamma dose rate measured 1 m from the canister centerline.

A typical canister is shown in Figure 22.2. This canister is approximately 30 cm in diameter by 3 m long and could contain 0.2 m^3 of 10-year old solidified waste from the processing of 3.33 tonnes of LWR fuel. Based on data from Section 2.4.3.1, the waste contained in a single canister of this size would have the following properties:

total fission product activity	= 7.72×10^5 Ci
total actinide activity	= 6.46×10^4 Ci
neutron generation rate	= 8.29×10^9 Ci
heat generation rate	= 4.40 kW.

FIGURE 22.2. High-Level Waste Canister

It may be desirable to seal the primary canister inside a second cylindrical stainless steel container whose top is welded closed. Potential advantages of overpacking are:

- Simplified decontamination. The secondary canister will not have been exposed to the contaminated process cell environment.
- Improved containment. The secondary canister will not have been subjected to the high temperatures and resultant thermal and mechanical stresses of processing operations associated with waste solidification.

Potential disadvantages of overpacking include:

- An increase in the cost of containment.
- An increase in the weight of the container for the same amount of material carried.
- A need to reduce the heat content of the waste because of a reduction in heat transfer properties of the container.

Studies of conceptual systems for the shipment of high-level waste^(6,7) indicate that several canisters of solidified waste could be shipped in a single steel, lead, or depleted uranium-shielded cask weighing approximately 100 tonnes. Such shipments would be made by rail. Smaller waste casks weighing approximately 22 tonnes could be shipped by truck, but, because of the smaller payload to cask ratio, these containers would be less efficient and more costly.

Cross sectional views of a conceptual cask^(6,7) are shown in Figures 22.3 and 22.4. The cask shown in these figures would be approximately 3 m in diameter by 3 m long. The canisters would be placed in a steel insert located within the cask cavity. The outer and inner cask liner would be fabricated from carbon steel. Lead or depleted uranium with a thickness of 20-30 cm would provide the gamma shielding required to reduce external dose rates to levels required by Federal regulations. The cask lid would be bolted to the cask body.

The high-level waste from processing a tonne of spent fuel that has been irradiated to about 25,000 MWd/MT may contain more than 200 g of ^{244}Cm . The neutron flux from spontaneous fission of this isotope requires that a hydrogenous material be included in the shielding of casks used for high-level waste shipments. Approximately 10 cm of borated water would provide the required attenuation of neutrons.

Circumferential fins would be used to promote natural convection and to provide a heat dissipation mechanism which maintains the outer surface of the cask below the temperature at which water in the neutron shield will boil. These fins would also provide an energy absorbing mechanism in the event of a transportation accident.

FIGURE 22.3. Conceptual High-Level Waste Shipping Cask

FIGURE 22.4. Cross Section of Shipping Cask

The number of canisters of solidified waste that may be transported in a single cask is subject to limitations. The temperature of the waste must not exceed the maximum temperature it received during solidification (approximately 1000°C), and, in the case of lead casks, the melting point of lead (327°C) must not be approached in the shield volume. Heat dissipation calculations^(6,7) carried out for a 100-MT cask with an insert diameter of 152 cm indicate that such a cask could accommodate thirty-six 15-cm diam waste cans, nine 30-cm diam cans, or four 60-cm diam cans of solidified high-level waste.

Transportation

Available modes of transport that will be used for the shipment of high-level waste include rail, truck, and barge. Air shipment is unlikely because of the weight of a typical shipping cask and the lack of any economic incentive to transport wastes quickly.

Most shipments of solidified high-level waste are expected to be made by rail because of previously stated advantages of shipping large casks. Truck shipments are technically feasible but economically unattractive when compared to rail shipments. Barge transport⁽¹⁾ may find limited application because it is economically attractive and because large casks can be shipped by this

mode. For barge transport to find wide application, reprocessing plants and waste repositories would need to be located adjacent to navigable waterways. Otherwise, it would be necessary to trans-ship waste containers from barge terminals to inland facilities.

22.4 TRANSPORTATION OF SPENT FUEL

22.4.1 Introduction

After irradiation in the reactor, nuclear fuel is usually cooled in a fuel storage pool for a minimum of 120 days. Subsequent to this cooling period it will be transported either to a reprocessing plant or to a fuel storage pool to await reprocessing or disposal (if it is decided not to reprocess spent fuel). The irradiated fuel assemblies contain unburned fuel, activated structural materials, transuranic elements and large amounts of highly radioactive fission products. The decay of the radioactive species produces large amounts of heat. Spent fuel shipping casks must therefore provide relatively large amounts of radiation shielding and have a large heat removal capacity. This section describes transportation systems currently available for shipping spent LWR fuel and presents information on anticipated future developments in the technology of spent fuel transportation.

22.4.2 General Transportation Requirements for Spent Fuel

The physical and radiological characteristics of spent nuclear fuel combine with the constraints of Federal transportation regulations to place stringent requirements on the transportation of spent fuel. (Federal transportation regulations are discussed in Appendix E.) Spent fuel assemblies from PWRs are 22 cm x 22 cm x 4 m long arrays containing 264 Zircaloy fuel pins about 1 cm in diam. The assembly contains about 460 kg of heavy metal and weighs about 670 kg. BWR fuel assemblies are about 14 cm x 14 cm x 4.5 m long arrays containing Zircaloy fuel pins approximately 1.4 cm in diam. A BWR assembly contains about 200 kg of heavy metal and weighs 280 kg. For this study a "reference" LWR fuel has been adopted. It is described in detail in Section 2.3. The characteristics that are important to transportation will be highlighted here. Properties of the reference fuel are summarized in Table 22.2.

After a 120-day cooling period the total radioactivity of the reference fuel assembly is 1.8×10^6 Ci. A spent fuel shipping cask uses γ shielding (typically lead or depleted uranium) to conform to the radiation dose requirements of 49 CFR 173.393. There is also a significant neutron flux produced from the spontaneous fission of ^{242}Cm and ^{244}Cm in the spent fuel. The neutron source strength from a reference fuel assembly is 1.3×10^9 neutrons/sec. A low atomic number material (typically hydrogen in water or a hydrocarbon) is used to shield the public and transportation workers from this neutron dose.

TABLE 22.2. Summary of Properties of a Reference LWR Fuel Assembly Cooled 120 Days (Extracted from Section 2.3)

Assembly total weight	454 Kg
Uranium/assembly	311 Kg
Total radioactivity	1.9×10^6 Ci
Fission Product Activity	1.7×10^6 Ci
Transuranic Content	5.5 Kg
Transuranic Activity	1.1×10^5 Ci
Neutron Source Strength	1.3×10^9 neutrons/sec
Total Thermal Power	8.4 kW
Total Gamma Power	1.4 kW
Exposure	25,000 MWd/MTU
Average Specific Power	35 MW/MTU

The reference assembly cooled for 120 days generates heat at the rate of 8.4 kW. Provision must be made in the spent fuel cask to remove this heat in order to protect the contents and the package from thermal damage and to remain below maximum temperatures specified in Federal regulations.

Spent fuel casks must satisfy the regulatory requirements for Type B packaging. They must maintain their shielding integrity and not release radioactive material if subjected to a hypothetical transportation accident environment. Provision must also be made to insure that a critical mass cannot be assembled in the cask during a transportation accident. The combination of fuel element length and the requirements for structural strength, gamma and neutron shielding and adequate heat removal characteristics make spent fuel shipping casks physically large and very massive. The number of assemblies that can be shipped at one time must be limited so that cask size and total weights remain within the weight and size limitations for truck and rail transportation.

Spent fuel casks must conform to handling requirements at the reactor fuel storage basin and at the fuel storage pool at the destination. Fuel is loaded and unloaded under water to provide radiation shielding to workers during these operations. The cask must be compatible with these under water operations. The spent fuel cask must have lifting points for the cask handling cranes at these facilities and its weight cannot exceed the capacity of the cranes. Casks must also be designed to facilitate decontamination before the loaded cask is shipped to the reprocessing plant and before the empty cask is returned to the reactor.

22.4.3 Available Transportation Systems

The technology to transport spent LWR fuel from current generation reactors exists today. Spent fuel casks are available that can be used to ship fuel with the maximum anticipated exposure after a minimum cooling time of 120 days. Casks are available for both truck and rail transport of spent LWR fuel. Most casks can accept either PWR or BWR fuel by using different fuel baskets, although a few casks are designed only for a particular fuel type. A list of available casks is presented in Table 22.3. A description of two of these casks is presented below. They are representative of the state-of-the-art in cask design and construction. The IF 300 cask is transported by rail and employs a water filled cavity and mechanical blowers for cooling. The TN8/TN9 casks are transported by truck. They have dry cavities and are cooled by natural convection.

- IF-300⁽⁸⁾ is a water cooled cylindrical cask designed to carry 7 PWR or 18 BWR fuel assemblies. Shielding is provided by water filling the cask cavity, 10.2 cm of depleted uranium metal in the cask shell and an exterior water-filled enclosure. Heat is removed from the fuel to the cask cavity walls by natural circulation of the contained water and transmitted through the cask sides and ends by conduction. The outer shell is surrounded by an annulus of water retained by a corrugated jacket. The corrugated jacket is cooled by two diesel-driven blowers with a total air moving capacity of 8.5 m³/sec. The maximum heat

TABLE 22.3. Available Shipping Casks for Current Generation LWR Spent Fuel

Common Name	Assemblies PWR/BWR	Shielding	Maximum Heat Removal (kW)	Approximate Loaded Cask Weight (MT)	Usual Transport Mode	Available Current/Under Construction
NFS-4	1/2 ^(b)	lead	11.5	24	truck	6/0 ^(c)
NFS-5	2/4	uranium	20	25	truck	0/0
IF-300	7/18	uranium	76	74	rail	4/0
TN-8	3/0	lead	35.5	36	overweight truck	0/2
TN-9	0/7	lead	24.5	36	overweight truck	0/3
TN-12 ^(a)	12/32	steel	~95	97	rail	0/1
NLI-1/2	1/2	lead	10.6	22	truck	3/2
NLI-10/24	10/24	lead	100	88	rail	0/12
BCL-6	1/1	lead	12.0	22	truck	0/1

a. Cask under construction for use in Europe. Licensing proceedings soon to be initiated in U.S. Heat removal is an estimate.

b. Means one (1) PWR assembly or two (2) BWR assemblies.

c. Means six (6) such casks are currently available and none (0) are under construction.

load is 76.9 kW corresponding to the heat generated from 7 PWR fuel elements with a burnup of 35,000 MWd/MTU at an average specific power of 40 MW/MTU after 120 days cooling. The loaded cask weighs about 61 tonnes. The cooling equipment weighs an additional 13.6 tonnes. The cask is normally transported by rail on a 100-tonne capacity flat-car. It may be transported for short distances on highways on a special 9-wheel truck. Such truck shipments require special overweight permits. The method is used primarily to move the cask from reactors without rail facilities to the nearest rail siding. A drawing of the IF-300 cask is presented in Figure 22.5.

- TN-8/TN-9⁽⁹⁾ The TN-8 is a cylindrical spent fuel cask employing the "dry" cooling method. The inner cavity is divided into three compartments each designed to carry a PWR fuel assembly with a maximum burnup of 38,500 MWd/MTHM and a minimum cooling time of 150 days. Gamma shielding is supplied by a layer of lead 185 mm thick sandwiched between the stainless steel inner cavity and the carbon steel outer shell. Together the lead and steel supply a total shielding equivalent of 230 mm of lead. The neutron shield consists of a 150-mm thick layer of resin affixed to the outer surface of the outer shell. The inner cavity of the cask is drained after the cask is removed from the fuel transfer pool. During transport heat is transferred from the fuel to the cavity walls mainly by radiation. Heat is dissipated to the ambient air by

FIGURE 22.5. The IF-300 Spent Fuel Cask Ready for Shipment⁽¹⁰⁾

means of a large number of nickel-coated copper fins welded to the outer shell and passing through the neutron shield. The maximum heat load of the cask is 35.5 kW. To provide impact protection end covers with balsa wood inserts are fitted to the cask before transport. The loaded cask weighs about 36 tonnes. It is transported by truck on a specially designed semitrailer. Overweight permits are required.

The TN-9 is similar to construction to the TN-8. The interior is divided into seven compartments each designed to carry a BWR fuel assembly with a maximum burnup of 36,500 MWd/MTU and a minimum cooling time of 150 days. The maximum heat removal capability is 24.4 kW. Drawings of the TN-8 and TN-9 casks are presented in Figures 22.6 and 22.7.

22.4.4 Transportation Systems Under Development

The technology to safely and efficiently transport irradiated LWR fuel is currently available. Casks designed and built in the near future will be replicas of or variations on currently available casks. The main thrust will be construction of a sufficient number of casks to meet the growing spent fuel transportation requirements of the LWR industry. The majority of spent fuel casks will continue to be moved by rail or truck. The choice of transport mode depends on economic factors and cask weights. Longer term developments in spent fuel cask design may include the following:

- Development of casks capable of transporting larger numbers of fuel assemblies.
- Development of casks for higher burnup, shorter cooling time fuel.
- Improvements in safety and handling features. Safety-related design efforts will mainly attempt to reduce cask weights while providing the same high degree of transportation safety.
- Improvements in neutron shielding to provide adequate shielding at decreased weights.
- Development of intermodal transportation systems utilizing barge transport.
- Development of casks with sufficient neutron shielding and heat removal capabilities to transport Pu recycle fuel, LMFBR fuel and HTGR fuel.

22.5 TRANSPORTATION OF PLUTONIUM

22.5.1 Introduction

Plutonium is separated from spent fuel at a reprocessing plant. The separated plutonium product is then shipped to a fuel fabrication plant for incorporation in mixed oxide fuel elements or to a facility for interim storage.

FIGURE 22.6. The TN-8 Spent Fuel Cask⁽⁹⁾

FIGURE 22.7. The TN-9 Spent Fuel Cask⁽⁹⁾

Current regulations permit the shipment of plutonium in either the liquid (plutonium nitrate) or the solid (plutonium oxide) form. After June, 1978, only shipments as a solid will be permitted. Shipment is permitted by truck, rail, or ship. There is currently a Congressional embargo on the shipment of plutonium by air. Currently most shipments of plutonium within the continental United States are made by truck. Shipments are made in containers certified by ERDA or by the Nuclear Regulatory Commission or specified in DOT regulations. All shipments must conform to Federal shipping regulations as outlined in Appendix E. Special precautions are taken to insure the security of plutonium in transit.

22.5.2 Considerations Important to Transportation

Regulatory Constraints

Federal regulations for the shipment of plutonium require that it be transported in packages which shall prevent loss or dispersal of the contents, retain shielding efficiency, assure nuclear criticality safety and provide adequate heat dissipation under both normal transport and hypothetical accident conditions. Shipments of plutonium in excess of 2 kg may be made only in accordance with Nuclear Regulatory Commission approved transportation plans which provide for the physical protection of special nuclear material in transit (10 CFR 73). All shipments must conform to Federal regulations for the transport of fissile radioactive material. (Federal transportation regulations are summarized in Appendix E.)

Regulations presently in effect allow the shipment of plutonium in either the solid or the liquid form. A recent revision of NRC regulations (10 CFR 71.42) specifies that after June 17, 1978, plutonium in excess of 20 Ci per package shall be shipped as a solid. The plutonium must be packaged in a separate inner container placed within outer packaging that meets the requirements for packaging of material in normal form. The separate inner container must not release plutonium when the entire package is subjected to the normal and accident test conditions described in Appendix A and Appendix B of 10 CFR 71.

Restrictions on air transport of plutonium, published in 10 CFR 73.32, specify that no shipment of plutonium in excess of 20 g or 20 Ci, whichever is less, shall be made in passenger aircraft. However, a recently enacted Federal statute prohibits all air shipments of plutonium until satisfactory demonstration has been made that shipping containers will maintain their structural integrity and not release their contents under conditions of high speed impact, explosion, or fire such as might be encountered in an aircraft accident. Design studies and tests of containers which would satisfy these aircraft accident damage criteria are currently underway at several laboratories under ERDA and NRC sponsorship.

Prior to 1969, there were no federal safeguards specifications for the transportation of special nuclear material (SNM). (See definition in Appendix E.) In 1969, a recognition of the potential for diversion of this material led to the issuance of 10 CFR 73 for the protection of SNM in transit. Revised security measures for the transportation of special nuclear material came into effect on March 6, 1974. The revised regulations apply to all shipments of plutonium in which the amount transported is equal to or greater than 2 kg.

Significant requirements contained in the March 1974 revised version of 10 CFR 73 are as follows:

1. Transportation of special nuclear material by common or contract carrier must be preplanned to insure that transit times are minimized and that routes are selected to avoid areas of natural disaster or civil disorder. An exchange of a hand-to-hand receipt is required at all points where there is a transfer of custody. Special nuclear material must be shipped in containers which are locked and sealed by tamper-indicating type seals. No container weighing 500 lb or less may be shipped on an open vehicle, such as an open truck or railway flatcar.
2. Shipments by road shall be made without any scheduled intermediate stops to transfer cargo between the point of origin and the destination. At least two people must ride in the transport vehicle. Transport vehicles must be equipped with a radiotelephone, and communication with a licensee or his agent must be made at predetermined intervals which normally do not exceed 2 hours. If the transport vehicle is specially designed to permit immobilization of the vehicle and to provide penetration barriers, an armed escort is not required. Otherwise, at least two armed guards riding in a separate vehicle must accompany the transport vehicle. The escort vehicle must be equipped for radio communication with the transport vehicle and for radiotelephone communication with the licensee or his agent. The cargo vehicle must be marked on top, sides and rear with identifying letters and numbers.
3. Shipments by rail must be escorted by two armed guards in the shipment car or an escort car of the train. The guards shall have the shipment under visual surveillance at all times. On-board radiotelephone communications capability must be provided with conventional telephone backup. Communication with the licensee or his agent shall be maintained as for shipments by road.
4. Shipments by sea must be made on vessels making the minimum ports of call. Transfer at domestic ports from other modes of transportation must be under continuous visual surveillance by a guard. There shall be no scheduled transfers from one ship to another. Shipments must be placed

in a secure compartment which is locked and sealed. Export shipments must be escorted by an authorized individual who may be a crew member, from the last port in the United States until it is unloaded at a foreign port. Ship-to-shore communications must be made every 24 hours to relay position information and the status of the shipment as determined by daily inspections.

5. Before a licensee can transport special nuclear material, a plan outlining procedures that would be used in meeting all the transportation requirements must be submitted to and approved by the Nuclear Regulatory Commission.

Design Constraints

The principal isotope of plutonium produced during fuel irradiation in a light water reactor is ^{239}Pu . Significant quantities of ^{238}Pu , ^{240}Pu , ^{241}Pu , and ^{242}Pu are also produced by neutron irradiation of LWR fuel rods. Table 22.4 summarizes the radioactive and decay heat properties of these isotopes of plutonium. Americium-241 is also included in the table because it is the daughter produced by beta decay of the relatively short-lived ^{241}Pu .

TABLE 22.4. Radioactive and Decay Heat Properties of Isotopes of Plutonium and of ^{241}Am

Isotope	Decay Mode	Half Life	Decay Energy (W/g)	Specific Activity (Ci/g)
^{238}Pu	α	87.8 y	5.70×10^{-1}	$1.75 \times 10^{+1}$
^{239}Pu	α	24,290 y	1.88×10^{-3}	6.14×10^{-2}
^{240}Pu	α	6,540 y	6.96×10^{-3}	2.27×10^{-1}
^{241}Pu	β	15 y	3.35×10^{-3}	$1.13 \times 10^{+2}$
^{242}Pu	α	387,000 y	1.13×10^{-4}	3.90×10^{-3}
^{241}Am	α	433 y	1.06×10^{-1}	3.24×10^0

In the design and fabrication of packages for the shipment of plutonium, problems of criticality, radioactive dose rate, and heat generation rate must all be considered. As can be seen from Table 22.4, the heat generation rate will depend primarily on the amount of ^{238}Pu present in the plutonium product and also to a more limited extent on the balance between ^{241}Pu and ^{241}Am .

22.5.3 Available Packaging and Transportation Systems

Packaging

Plutonium is classified in Transport Group I. (See definition in Appendix E and in 49 CFR 173.389-0.390.) For radionuclides in this transport group, not more than 1 mCi of activity can be shipped in Type A packaging. A shipment containing more than 1 mCi of plutonium must be made in Type B

packaging. Plutonium is a fissile material, so all plutonium shipping containers must conform to the additional requirements of packaging for fissile material.

The following containers meet the requirements for Type B packaging for fissile materials:

1. Specification 6M metal packaging (see 49 CFR 178.104). Specification 6M containers are authorized only for solid radioactive materials which will not decompose at temperatures up to 121°C (250°F). Radioactive thermal decay energy of the package contents may not exceed 10 W. The radioactive material must be packaged in one or more sealed and leak tight metal cans or polyethylene bottles within a Specification 2R (see 49 CFR 178.34) containment vessel. The 6M is the most commonly used specification container for the shipment of plutonium in the solid (oxide) form.
2. Certified packaging. Any other Type B packaging which also meets the standards for packaging of fissile radioactive materials in the regulations contained in 10 CFR 71 and for which a certificate of compliance is issued by the U.S. Nuclear Regulatory Commission or by ERDA.
3. Any other Type B packaging which also meets the pertinent requirements for fissile radioactive materials in the current regulations of the International Atomic Energy Agency and for which the foreign competent authority certificate has been revalidated by the U.S. Department of Transportation. [See 49 CFR 173.396(b)(5).]

A description of the 6M container is given below. Another container which has been used for plutonium shipment, the LLD-1, is also described. Additional information on these containers is found in Reference 11.

DOT Specification 6M Container

A typical specification 6M container of 10-gal capacity is shown in Figure 22.8

Authorized contents: Up to 4.5 kg of plutonium metal, alloy or compound or up to 13.5 kg of ²³⁹U metal or alloy. The material must be in the form of a solid which will not decompose at temperatures up to 250°F. Additional Fissile Class restrictions and restrictions on ratios of hydrogen to fissile material are given in 49 CFR 173.396.

Description of outer container: The outer steel drum must have a rated capacity of not less than 10 gal nor more than 110 gal and must conform to specification 6C or 17C steel drum as defined in 49 CFR 178.99 or 49 CFR 178.115. Each drum must have at least four 1.2 cm (0.5 in.) diam vent holes, located on the sides of the drum, near the top, each covered with a weather-proof tape, fusible plug, or equivalent device. The inner containment vessel

FIGURE 22.8. Specification 6M Container

must be centered within the outer shell by means of solid disks or rings made of industrial cane fiberboard or hardwood or plywood. The sides of the inner vessel must be protected by at least 9.5 cm (3.75 in.) of insulation media such as vermiculite or celotex.

Description of inner container: The inner container must conform to specification 2R or equivalent (except that only carbon steel or stainless steel is authorized) with a maximum usable inside diameter of 13.3 cm (5.25 in.), minimum usable inside diameter of 10 cm (4 in.) and minimum height of 15 cm (6 in.). The plutonium must be packaged within one or more sealed and leak tight metal cans or polyethylene bottles within the 2R inner containment vessel.

Shielding: None provided. May be added within the containment vessel when required.

Heat removal capacity: No design features for cooling. Normal licensed limit of 10 W. A 10-W source results in a containment vessel temperature of 155°F for a 70°F ambient temperature.

LLD-1 Container

The following summary of the LLD-1 package is taken from Reference 1. Additional details of this package as used for shipment of special nuclear material are contained in a Safety Analysis Report prepared by E. I. DuPont de Nemours and Company.⁽¹²⁾ The LLD-1 package is shown in Figure 22.9.

Authorized contents: Not more than 4.5 kg of plutonium metal or oxide, or Pu-U alloy containing not more than 4.5 kg of fissile material, any enrichment of uranium.

Description of outer container: A cylindrical outer container is supported in a cage of 3/4 in. O.D. x 1/16 in. wall seamless steel tubing. The outer container (bird) is fabricated from 1/4 in. seamless steel tubing or forging. It is closed by engaging the closing lugs on the bird. Seal is affected by tightening the lid against a gas-filled O-ring using the hex head bolts on the lid assembly. For transporting, the cage is placed inside a sheet metal box 16.5 in. x 16.5 in. x 26 in. high.

FIGURE 22.9. LLD-1 Container

Description of inner container: The inner container is a DOT-2R container, fabricated of 5-in. Sch. 80 steel pipe with a threaded plug and a gas-filled O-ring seal. Oxide powder is contained in two No. 8 steel cans which are placed inside the inner container. The inner container is lined with padding to minimize damage to the steel cans during transport.

Shielding: None provided.

Heat removal capacity: By convection only. The package can accommodate a heat generation rate of approximately 20 W.

Transportation

Most shipments of plutonium in the continental United States are made by truck. Shipments of large quantities of government-owned plutonium which originate or terminate at ERDA laboratories or ERDA prime contractors are usually made in ERDA courier-driver vans supervised by the Albuquerque Operations Office. Commercial shipments of plutonium are normally made by common carrier. Where large quantities of plutonium are involved, commercial shipments are made in exclusive use vehicles and are accompanied by an escort vehicle. All shipments conform to regulations for the transport of fissile radioactive material as specified in 49 CFR 173.389-399 and 10 CFR 71, and to regulations for the protection of special nuclear material in transit as specified in 10 CFR 73 and associated regulatory guides. NRC-approved transportation plans which describe how the performance requirements of Part 73 will be met are required of all shippers of plutonium.

Several factors limit the number and arrangement of packages of plutonium which can be transported in a single shipment. Depending on the fissile classification of the package, the number of packages that can be shipped at one time may be limited to prevent an accidental criticality during transportation. For Fissile Class II shipments, the number of packages placed in one vehicle is limited to an aggregate transport index of 50. The radiation dose limits during transportation (see Appendix E) may also be a limiting factor when unshielded packages are used.

22.5.4 Packaging and Transportation Systems Under Development

Packaging

Typical packages which are currently used for the shipment of plutonium are described in the section on Available Packaging and Transportation Systems (Section 22.5.3). These packages have a limited carrying capacity, a low heat dissipation capability, and no shielding. They have proven adequate for the transport of approximately 5 kg quantities of plutonium generated at relatively low exposure (approximately 90 to 95% ^{239}Pu and up to 10% ^{240}Pu). However, the plutonium which will be packaged for shipment during the next

3 decades will require a substantially greater carrying capacity, more gamma and neutron shielding, and a greater heat dissipation capacity than is available in present packages.

A package for the shipment of 32 kg of plutonium oxide is currently being developed by Allied-General Nuclear Services and Battelle-Columbus Laboratories.⁽¹³⁾ The isotopic composition of plutonium which is being assumed as a design basis for the package is given in Table 22.5.

TABLE 22.5. Design Basis Isotopics for
32 kg Plutonium Package

Isotope	Abundance
^{238}Pu	2.5 wt%
^{239}Pu	50 wt%
^{240}Pu	25 wt%
^{241}Pu	15 wt%
^{242}Pu	7.5 wt%

The isotopic composition shown in Table 22.5 is approximately that of second generation recycle plutonium. The key isotope is ^{238}Pu which is responsible for more than 80% of the design basis decay heat and a large fraction of the fast neutron source. Thirty-two kilograms of plutonium with the foregoing isotopic composition will generate about 640 W of decay heat and produce an unshielded radiation source intensity great enough to generate contact dose rates in excess of 2 rem/hr.

The design objectives for the PuO_2 packaging are as follows:

- A minimum capacity of 25 kg of Pu per package.
- Dimensional outside envelope which approximates the dimension of the L-10 container.
- Packaging components also to be useful for storage of PuO_2 .
- Fissile Class I configuration.
- Shielding adequate to restrict facility operations personnel to 0.5 rem/year/man package assembly exposure.
- Meet all pertinent requirements of 10 CFR 71 and 49 CFR 173.389-399.

A package which is expected to meet these design criteria is shown in Figure 22.10. The package consists of:

- four PuO_2 powder canisters containing a total of 32 kg of PuO_2
- a primary pressure vessel for packaging and/or storage
- a Type B overpack.

FIGURE 22.10. Plutonium Products Package

Details of the canisters and the primary pressure vessel are shown in Figure 22.11.

Eight kilograms of plutonium oxide powder can be contained in each 15.2 cm diameter x 30.5 cm high thin-walled and vented stainless steel canister. The canisters are designed with an annular powder volume to limit centerline temperatures. The vent on each canister permits the escape of water vapor and radiolytically generated gases while providing particulate filtration to contain the oxide powder.

The primary pressure vessel will contain four canisters. This pressure vessel, when contained in the Type B overpack, is designed to withstand rough handling and transportation accident conditions without loss of containment. A vent is provided to permit controlled release of gas pressure either prior to shipping or after receipt. However, the pressure vessel is designed to withstand pressures resulting from the combined effects of total radiolytic decomposition of moisture adsorbed on the oxide powder and accident condition temperature.

FIGURE 22.11. Primary Pressure Vessel and Canister for Plutonium Products Package

Figure 22.10 shows the pressure vessel and its canisters with contents being transported in a Type B overpack. The overpack pressure vessel is designed to the standards of Section VIII of the ASME Boiler and Pressure Vessel Code and it provides containment for gases and particulates should the canister and primary pressure vessel fail for any reason. Radiation shielding is provided by a combination of the heavy-walled pressure vessels and an 20.3-cm thick neutron shield annulus. A fire protection shell is fit over the outer neutron shield containment shell. The fire protection shell expands away from the neutron shield containment shell when subjected to a fire environment thus providing a significant thermal barrier.

The overpack outer dimensions are 76.2 cm in diameter and 183 cm high. The total package (plutonium oxide powder, canisters, primary pressure vessel and overpack) will weigh approximately 1360 kg. A safety analysis report for the package is currently being prepared.

Transportation

Concern for the possibility of diversion of special nuclear material has resulted in an emphasis on safeguards procedures. Two research programs currently being carried out under ERDA sponsorship will increase the protection of special nuclear material in transit.⁽¹⁴⁻¹⁷⁾ One program involves the design and testing of a safe-secure transport vehicle which will provide protection from armed attack and will incorporate vehicle immobilization and penetration resistance features. The second program involves the design and testing of an improved communications system which will provide continuous communication with special nuclear material shipments.

The design goals for specially designed safe-secure cargo vehicles are listed in Table 22.6.

TABLE 22.6. Design Goals for Specially Designed Cargo Vehicles

<u>Feature</u>	<u>Strategy</u>
1. Immobilization System.	1. Provide controlled stop, resist release.
2. Access Denial.	2. Provide a variety of barriers to resist physical penetration. Intrusion penalties could provide additional deterrence.
3. Driver Protection	3. Provide an armored cab to protect occupants.
4. Communication System.	4. Provide essentially continuous communication with licensee or agent.

Immobilization of the transport vehicle can be accomplished by one of three methods:

1. locking of wheels
2. disabling the engine and/or the drive line
3. driving the steering system to an extreme position and securing it in that position.

Of the three systems listed above, the wheel locking system appears to be the most desirable from the standpoint of safety, cost and reliability.

Access denial includes both penetration deterrence and intrusion penalties. Penetration deterrence involves the construction of a vehicle which presents the highest possible degree of difficulty to unauthorized individuals seeking to enter the cargo compartment by forcible entry. Penetration deterrence includes the use of special hinges and locking devices on doors and the fabrication of laminated and armor plated cargo compartment walls. Intrusion penalties involve the use of devices which attack an intruders senses and make it difficult or impossible for him to function without the aid of protection devices.

Driver protection is afforded by cab armor which shields the occupants from the initial phases of an attack and allows time for initiation of immobilization and deterrent systems and for communication with security forces on the outside.

Trucks employing some of the features described above have been constructed for use in the shipment of special nuclear material to and from ERDA Laboratories. These trucks are under the supervision of the Albuquerque Operations Office of ERDA.

Current regulations (10 CFR 73.31) for shipment of plutonium by truck require the equipping of cargo and escort vehicles with radiotelephones and the reporting of position at least every 2 hours, or within 5 hours by fixed telephone if radiotelephone contact cannot be made. Continuous capability for communication between a transport vehicle and the licensee or his agent while the vehicle is enroute is not possible with radiotelephones.

A communications system currently under study and development⁽¹⁷⁾ has a goal of continuous communication between a transport vehicle and its carrier dispatcher. The generalized system concept for this communications network is shown in the diagram of Figure 22.12. The system involves message transmission in digital code by high frequency radio between the vehicle and a remote terminal. Remote terminals are connected by leased line to the central terminal of the communications network. The central terminal contains equipment for recording the position and status of each transport vehicle on a continuous basis.

FIGURE 22.12. Conceptual System for Continuous Communication Between Plutonium Transport Vehicles and Carrier Dispatchers

Messages are transmitted by commercial telephone circuits between the central terminal and individual carrier dispatchers. The central station also summons assistance from the appropriate law enforcement agencies in the event of an emergency.

22.6 TRANSPORTATION OF CLADDING HULLS

22.6.1 Introduction

In current fuel reprocessing operations fuel bundles are sheared in short lengths and the fuel pellets are acid leached from the cladding. The residue from the acid leach includes short lengths of fuel cladding with some residual fuel, massive end fittings, fuel support grids and assorted springs, spacer elements and fuel bundle support rods.

Two methods have been used for disposal of the cladding waste. The West Valley Reprocessing Plant operated by Nuclear Fuel Services disposed of the hulls by onsite burial. Onsite burial in retrievable containers is expected to be used at Allied-General's Barnwell Plant. The Windscale Plant in Britain places the cladding waste in an onsite underwater storage facility. A similar method was to be used at General Electric's Midwest Plant.⁽¹⁸⁾ In the future cladding hulls may be shipped offsite to a disposal or treatment facility. This material may also receive some treatment onsite before being transported. This section describes transportation systems currently available and those under development for the transportation of chop-leach cladding hulls and fuel bundle hardware.

22.6.2 Properties of Untreated Waste Important in Transportation

The physical and radiological properties of cladding hulls and fuel bundle hardware from current generation Light-Water Reactors are not well known.⁽¹⁸⁾ These properties will vary with reactor type (BWR or PWR) and fuel element manufacturer. For this study a "reference" LWR fuel has been adopted. The properties of the cladding hulls and hardware from the reference fuel are outlined in Section 2.4. Additional information on cladding hull wastes is contained in Section 7. The properties that are important to transportation of these materials will be discussed briefly here.

Approximately 325 kg of cladding hulls and fuel bundle hardware is generated for each tonne of fuel reprocessed. This is comprised of about 228 kg of Zircaloy, 29 kg of stainless steel and 8 kg of Inconel. The untreated waste occupies a volume of about 0.32 m³.

The radiation and heat generation properties of the cladding and hardware are the most important parameters in selection or design of a shipping container for transportation of these materials. There is neutron-induced radioactivity in all of the fuel bundle materials removed from the reactor. Significant activity is also associated with the fuel that remains with the

cladding after the acid-leach process. These activity levels have not been extensively measured for current generation fuel. Estimates of these activity levels are based on computer calculations with the ORIGEN Code. The few measurements that have been performed roughly confirm the computer calculations. (See Section 7.1.)

The radiation and heat generation rates of the reference LWR fuel hulls and hardware with 0.05% residual fuel are presented in Table 2.14. One year after reactor discharge the total activity of the cladding waste from 1 MTHM is 3.6×10^4 Ci and the total heat generation rate is about 195 W. A neutron dose that could be significant in determining transportation shielding requirements is produced by the ^{242}Cm and ^{244}Cm in the residual fuel. The neutron source strength is about 2×10^6 neutrons/sec for the clad from 1 MTHM. Total contamination from transuranic elements is about 5.3×10^5 nCi/g of Zircaloy after a 1-year decay period. Zircaloy will also contain transuranic isotopes produced in the reactor from ASTM allowed maximum uranium impurity level of 3.5 ppm. Typical uranium impurity levels are about 1 ppm. This has been calculated to produce about 90 nCi (mostly ^{241}Pu) TRU elements per gram of cladding irradiated for 33,000 MWd/MTU in a flux of 2.9×10^{13} neutrons/cm²-sec.⁽¹⁹⁾ NRC has proposed rulemaking that would designate all Zircaloy from high exposure LWR fuel as TRU waste.

The physical and radiation properties of the reference LWR fuel cladding hulls and hardware with 0.05% residual fuel are summarized in Table 22.7. There is evidence that micron-sized particles of zirconium (usually called zirconium fines) are pyrophoric. Zirconium fines are expected to be produced in the chop-leach operations at the reprocessing plant. The self-ignition temperature of the fines depends on particle size and the amount of fines

TABLE 22.7. Properties of Reference LWR Fuel Cladding Hulls and Hardware from 1 MT Fuel 1 Year After Reactor Discharge

Mass	
Zircaloy	288 kg
Stainless Steel	29 kg
Inconel	8 kg
Average Density (Untreated)	1 g/cm ³
Total Radioactivity	3.6×10^4 Ci
Heat Generation Rate	195 W
Neutron Source Strength	2×10^6 neutrons/sec
TRU Contamination	
External (from fuel)	5.3×10^5 nCi/g
Internal (from U impurity)	90 nCi/g

present. One kilogram of 1 μm -sized particles may be pyrophoric at 60°C. It may be necessary to separate the fines from the remainder of the cladding waste before it is stored or transported.

22.6.3 General Transportation Requirements for Untreated Hulls

Cladding hulls and hardware must be transported in compliance with applicable Federal regulations. These regulations are summarized in Appendix E. Cladding hulls must be shipped in Type B packaging with adequate shielding to conform with the radiation dose requirements of 49 CFR 173.393. The package must also provide adequate heat removal capabilities. The shielding thickness and heat removal requirements of the packages will depend on the quantity of material shipped and on the cooling time that it has received. Conservative calculations have determined that the hulls and hardware from 5 MTU of high burnup PWR fuel transported 6 months after reactor discharge would require a shipping container with about 18 cm of lead shielding.⁽²⁰⁾

22.6.4 Effects of Treatment Options on Transportation Requirements

A variety of treatment options have been identified for cladding hulls. These options are discussed in detail in Section 7. Most of the treatments proposed for cladding hulls are in the preliminary stages of development. Many are only conceptual at this time and none of the technologies have been applied to current generation cladding hulls on a commercial scale. It is not clear which, if any, of these alternatives will be eventually selected for the treatment of cladding hulls. The treatments might also be applied in various combinations. The hulls could be treated at the reprocessing plant and/or at a central treatment plant handling hulls from several reprocessing plants. For these reasons the effects of the various treatment options on transportation requirements can be discussed in only a general way. The effects of selected treatments on transportation requirements are presented below:

- Interim Onsite Storage. The hulls and hardware may be stored onsite at the reprocessing plant before being transported to a treatment or disposal facility. The total activity of the hulls and hardware decreases by more than a factor of 10 after a 10-year cooling time (see Table 2.14). The heat generation rate is reduced by a similar amount. The shielding and heat removal requirements of the shipping containers could be substantially reduced by onsite storage for 10 years.
- Sorting. The stainless steel end fittings and Inconel spacers may be mechanically separated from the cladding hulls. The activity and heat generation rate of the Zircaloy hulls decrease rapidly with cooling time, leveling out at approximately 2 years after reactor discharge. (See Figure 7.2.) Segregation of the end fittings and spacers would produce two waste streams. The end fittings and spacers comprise a small

percentage of the waste volume, but contain the majority of the long-lived radioactivity requiring shielding. The hulls form a relatively low-level high-volume stream after a suitable cooling period.

- Chemical Decontamination. Most of the residual fuel on the cladding hulls can be removed by appropriate chemical treatments. The neutron dose from the hulls is due in large part to the curium in the residual fuel. Chemical decontamination would reduce the neutron shielding requirements of the shipping container. The gamma dose is produced primarily from the clad itself. Gamma shielding requirements and heat removal capabilities would not be appreciably changed by chemical decontamination.
- Densification. Melt densification of the cladding hulls may decrease waste volumes by about a factor of 6. Mechanical compaction could increase the density of the waste by a factor of 4 to 5.⁽¹¹⁾ Densification of the hulls would increase the radioactivity per unit volume, tending to increase shielding requirements. This effect is mitigated by an increase in self shielding. The net effect on transportation shielding requirements has not been determined. Overall payload to total shipping weight ratios should be increased by densification of the waste.
- Matrix Formation. A matrix material such as sand, bitumen, or concrete may be added to the waste sometime before shipment to prevent generation of pyrophoric zirconium fines during transportation. Addition of a matrix material would increase the weight and, to a lesser extent, the volume of the material to be transported. Some decrease in shielding requirements would result from the shielding provided by the matrix material.
- Chemical Conversion. Some chemical conversion processes that have been proposed would use the converted hulls as a high-level waste absorbent. (See Section 7.) The hulls would then be transported with the high-level waste. If the converted hulls are not incorporated in the high-level waste, they will require transportation to a disposal site. Transportation requirements for chemically converted hulls are largely the same as for untreated hulls. The primary effect of chemical conversion from a transportation and packaging viewpoint is a modest decrease in waste volume.
- Volatilization. Volatilization processes produce two waste streams. The major volume stream will probably be a non-TRU, low-level solid. Transportation of low-level solids is described in Section 22.7. The minor volume stream contains the bulk of the TRU elements and other radioactive materials. It would require a shipping container with appropriate shielding and heat removal characteristics for transportation to a disposal site.

- Burning. Some volume reduction occurs when the hulls and stainless steel are burned. The Inconel does not burn. It could be separated and handled independently or left with the oxidized metal. Except for the effects of volume reduction discussed previously, technologies using burning have only minor effects on transportation requirements.

22.6.5 Available Transportation Systems

To date, cladding hulls and hardware have been transferred to containers at the completion of the chop-leach process and buried or stored onsite at the fuel reprocessing plant. Consequently, there are no shipping containers presently available which are designed specifically for the transportation of the hulls and hardware. Some radioactive material packaging is available that might be used to transport the hulls. Prior to shipment, it would probably be necessary to transfer the hulls and hardware from the existing hull disposal containers to a container designed specifically for the shipping cask used. Some modifications to the cask might also be required. The hull containers used must also be compatible with handling and storage requirements at the destination. Presently available shipping casks are transported by truck or rail. The choice of transport mode is generally determined by the size of the cask and economic considerations.

A list of some presently available casks that might be used to transport cladding hulls and hardware is presented in Table 22.8. Since the shipping form for the cladding hulls has not been determined, the table lists a wide variety of available Type B packaging. Many of the containers would not be suitable for untreated hulls because of inadequate shielding or small volume. These or similar containers might be used for some of the treatment products discussed previously. It may also be possible to add shielding to some of the containers or otherwise modify them to allow their use for a particular waste form. Some of the casks, like the GE IF100 and IF200, have greater heat removal capabilities and larger amounts of shielding than will probably be required for cladding hulls. They provide a safe method for transporting the hulls, but some unnecessary weight is transported, resulting in an economic penalty to the shipper.

Table 22.8 is not a comprehensive list of available Type B packaging, but it does indicate the variety of such containers that have been built. Containers are available that could be used to transport untreated waste or cladding treatment products, although the particular container that would meet the regulatory requirements to transport the waste safely may not be well-suited economically.

TABLE 22.8. Type B Packaging for Possible Use in Transportation of Cladding Hull Wastes

Container Name	Cavity Dimensions (Volume)	Shielding	Heat Removal Capacity	Approximate Weight (MT)	Usual Mode of Transport
ATMX-600 Railcar ⁽¹¹⁾	2.7 m x 2.7 m x 15.2 m (111 m ³)	None	500 watts	49.0 (net)	Rail
Super Tiger ⁽¹¹⁾	1.9 m x 1.9 m x 4.37 m (16.3 m ³)	None	235 watts	13.6 (net)	Truck, Rail
ORNL Model III ⁽²¹⁾	73 cm D x 1.98 mH (85 m ³)	23 cm steel	440 watts	19.1	Rail
SRP ⁶⁰ Co	38 cm x 58 cm x 93 cm (0.22 m ³)	22 cm lead	4200 watts	14.3	Truck
BMI-1 ⁽¹¹⁾	39 cm D x 1.37 mH (0.16 m ³)	20 cm lead	1500 watts	10.7	Truck
Paducah Demon- stration Cask	43 cm D x 1.29 mH (0.19 m ³)	15 cm uranium	10,500 watts	13.6	Truck
NCL 6502	30 cm x 30 cm x 3.3 m (0.3 m ³)	25 cm lead	7,300 watts	Unknown	Truck
GE IF200B140 ⁽²¹⁾	38 cm D x 3.55 mH (0.4 m ³)	21 cm lead	5,800 watts	25.4	Truck
GE IF100X137 ⁽²¹⁾	33 cm D x 3.47 mH (0.3 m ³)	21 cm lead	5,800 watts	20.4	Truck
GE IF100B129 ⁽²¹⁾	33 cm D x 3.27 mH (0.28 m ³)	21 cm lead	13,000 watts	20.0	Truck
NECO B3-1	66 cm D x 1.06 mH (0.37 m ³)	15 cm lead	Unknown	9.3	Truck
ATCOR LL-50-100	1.6 m D x 1.9 mH (3.71 m ³)	11.4 cm lead	Unknown	24.0	Truck
ATCOR LL-57-65	0.91 m D x 2.9 mH (1.93 m ³)	17.8 cm lead	Unknown	25.8	Truck
ATCOR BC-48-220	1.85 m D x 2.23 m H (6.04 m ³)	17.8 cm concrete and 6.4 cm steel	Unknown	21.8	Truck
Brookhaven Waste Shipment Overpack (Two Sizes)	0.58 m x 1.6 m x 0.35 m (0.32 m ³)	43 cm concrete	Unknown	13.6	Truck, Rail
	1.1 m x 0.6 m x 1.8 m (1.2 m ³)	30 cm concrete	Unknown	13.6	Truck Rail

22.6.6 Transportation Systems Under Development

Although some presently available shipping containers might be used to transport cladding hulls and hardware, a cask designed specifically for shipping this material would be desirable for several reasons:

1. It could be designed with the optimum shielding thickness and cavity dimensions to give the best payload to weight ratio.
2. The design would insure adequate heat removal characteristics.
3. Cask dimensions could be designed to conform with cladding hull handling equipment already in use at some reprocessing plants.
4. There would be no problem with availability of the casks, since they would only be used for cladding hull transportation and could be owned or leased by the reprocessing plant operator.

The technology exists today to design and construct a cask specifically for cladding hull transportation. The development of such a cask requires an accurate description of the waste to establish design parameters and an appropriate testing program to satisfy licensing requirements.

Some preliminary work has already been carried out for casks designed specifically for the shipment of cladding hulls. Blomeke and Perona⁽²²⁾ investigated the use of 22-cm and 61-cm diam ASTM A-53 steel pipe-to-can compacted cladding hulls. Their analysis used a computer code to design a finned, cylindrical cask to carry these cans. They investigated the economic effects of various can and cask configurations, concluding that the most economical feasible cask was about 3.6 m long, 1.8 m in diameter and carried twenty-seven 22-cm diam cans of compacted cladding cooled for 1 year after reactor discharge. The cask would accommodate the cladding from about 9 days of reprocessing activities. About 14 cm of lead shielding would be provided and the cask would weigh about 81 tonnes. Rail transportation would be required because of the cask weight. The activity in the cladding was calculated using the ORIGEN computer code, assuming exposure in a PWR with a burnup of 33,000 MWd/MTU and a specific power of 30 MW/MTU. It was also assumed that 0.5% of the fission products and 0.1% of the actinides originally in the fuel remained with the cladding.

A more detailed study has been undertaken by Allied-General Nuclear Services⁽²⁰⁾ to design a cask for use at their Barnwell Nuclear Fuels Plant. Their analysis is conservative, but probably representative of cask designs that will be developed in the near future. Their calculations are based on PWR fuel with a burnup of 40,000 MWd/MTU at a specific power of 35 MW/MTU in a flux of 2.74×10^{13} neutrons/cm²-sec. They assumed that 0.5% of the fuel remained with the cladding. Radioactivity levels are based on calculations with the ORIGEN computer code. Their cask was designed to be used with the cladding hull

handling equipment presently installed at their plant. The untreated cladding hulls, spacers and end fittings would be placed in a 1.1 m diam by 2.4 m long container constructed of 0.3 cm stainless steel. The bottom of the container is dished and reinforced. The container is closed by bolting on a 2.5-cm thick top piece. The top is vented to prevent pressure buildup from trapped water or hydrogen generation. This vent is capped before the container is shipped. The option exists to add a sand matrix to the can, but it is not expected to be necessary.

For shipment, the cans containing the clad from 1 day's processing (5 tonnes of fuel), are placed in a finned, cylindrical cask with a 2.5-cm thick steel inner liner, 17 cm of lead shielding and a 5-cm outer steel shell. The cask would weigh about 100 tonnes and would require rail transportation. Shielding thicknesses are based on a conservative 5 mrem/hr (1/2 the value allowed by Federal regulations) at 6 ft from the transport vehicle for hulls shipped 6 months after reactor discharge. No allowance is made for self-shielding. Calculations for hulls shipped 1 year after reactor discharge indicate that the shield thickness could be reduced about 1.9 cm of lead if self-shielding effects are included.

The AGNS study establishes design parameters for a cask to ship untreated cladding hulls with no interim onsite storage. Most cladding hull treatment products are expected to require shipping containers with less stringent design parameters. No technical obstacles to the design and construction of a cask for untreated hulls has surfaced as a result of the AGNS study. The biggest difficulty appears to be a lack of good experimental data to characterize the physical and radiological properties of the untreated cladding. The conservative approach required because of this lack of information results in a cask that will probably have shielding and heat removal capabilities exceeding actual requirements. Cask requirements are also highly dependent on the treatment options selected. AGNS has taken a "worst case" approach, but firm cask designs must await decisions on the selection of one or more of these treatment options.

The AGNS cask is designed to suit their particular needs. There are no technical reasons why casks of different dimensions could not be designed to meet the needs of other reprocessing plants or other shippers of cladding hulls. For example, design of a scaled down version of the AGNS cask that could be transported by truck should be straightforward. Many cask design parameters will be set by handling requirements of the shipper and economic considerations within the framework of NRC regulations.

22.7 TRANSPORTATION OF NON-HIGH-LEVEL WASTE

22.7.1 Introduction

This section will describe in a general way the packaging and transportation modes that are available or might be developed to ship the wide variety of materials included in the classification of non-high-level waste. Non-high-level wastes are generated at all nuclear facilities. They include an assortment of materials with a wide range of radioactivity levels. Non-high-level wastes include such items as resins and filter cartridges from liquid cleanup systems, HEPA filters, contaminated failed or obsolete equipment, radioactive gases and a variety of contaminated trash such as plastic bags, gloves, protective clothing and other particles of wood, paper, plastic and rubber. Some of the waste is contaminated with transuranic elements.

On a volume basis, the majority of the fuel cycle wastes requiring transportation are included in the general category of non-high-level waste. There are a multitude of treatment steps that can be applied to these wastes before they are shipped. Some of these treatments can significantly affect packaging requirements.

Most shipments of non-high-level waste are made by truck. Some shipments travel by rail. This trend is expected to continue, with the choice of transport mode dependent on convenience and economic considerations.

22.7.2 Properties of Untreated Wastes

The properties of untreated non-high-level wastes from reactors, fuel storage facilities, reprocessing plants and fuel fabrication facilities are discussed in some detail in Section 2. The properties that are important to transportation will be outlined here. In line with recent trends,⁽²³⁾ it is assumed that all liquids will be fixed in a solid before being transported.

- Reactor wastes are primarily from the water treatment systems. The characteristics of the waste vary with reactor type and the particular cleanup method used. The majority of the reactor waste volume is contaminated resins. Contamination levels vary from 0.1 to 650 Ci/m³. These wastes are combustible solids contaminated with mixed fission products. Contaminated filter cartridges from the water treatment system range from 0.04-1785 Ci/m³ with typical values of about 85 Ci/m³. These wastes are also combustible solids contaminated with mixed fission products. Other reactor wastes include an assortment of HEPA filters, gloves, protective clothing and contaminated failed equipment. Contamination levels can vary widely, with an average of about 0.08 Ci/m³. Nearly all shipments of radioactive wastes from reactors involve Type A quantities of radioactive material. Reactor wastes are presently disposed of at a commercial burial ground.

- Reprocessing plant wastes include a wide variety of TRU and non-TRU wastes. The most highly radioactive wastes will be failed equipment from the head-end and co-decontamination processes. Removed HEPA filters may also have high radiation levels. Other wastes that will require shielding during transportation include small amounts of wet and dry solids, and relatively large amounts of combustible and noncombustible trash. Waste not requiring shielding during transportation includes failed equipment, filters, combustible and noncombustible trash and other dry solids. Wastes from head-end and co-decontamination processes and plutonium separation and finishing operations are expected to be contaminated with TRU elements. If the NRC proposed TRU rulemaking is adopted, these wastes will be transported to a Federal repository. Waste from the remainder of the plant will be contaminated with mixed fission products and may be disposed of at a commercial burial ground.

Radioactive gases are also generated at reprocessing plants. The most important of these are iodine, krypton and tritium. Under present regulations krypton and tritium are dispersed to the atmosphere. Future requirements may necessitate collection and disposal of these gases. Iodine will be trapped in some solid medium for disposal. Tritium will probably be chemically fixed in a solid. Krypton alone or with other noble gases, will probably be placed in pressurized cylinders for shipment to a storage facility. (See Sections 13 and 14.)

- Mixed oxide fuel fabrication plant wastes are all potentially contaminated with transuranic elements. They include combustible and noncombustible solids with an average contamination level of 1000 Ci/m^3 , HEPA filters contaminated to 4000 Ci/m^3 and failed equipment and glove boxes with a wide range of contamination levels. Although contamination levels appear high in mixed oxide plant wastes, much of the activity is alpha and beta radiation and will require little shielding.
- Fuel storage pools may be built independently of reactors and fuel reprocessing plants. Wastes from these facilities are expected to be Type A quantities of non-TRU waste from pool water filtration systems. They will include spent resins, filter cartridges, HEPA filters and solidified liquids.

22.7.3 Regulatory and Other Constraints

All shipments of radioactive materials must be made in compliance with the regulations of the U.S. Department of Transportation and the Nuclear Regulatory Commission. These regulations are discussed in Appendix E. The packaging requirements for radioactive materials that are set by the regulations vary with the type and quantity of radioactive material to be shipped.

Some shipments of non-high-level waste qualify as low specific activity material, requiring only strong, tight industrial packaging for transportation. Other shipments require Type A packaging and some require properly shielded Type B packaging.

The facility receiving non-high-level waste can impose requirements on the waste packaging in addition to the constraints of Federal transportation regulations and the physical nature of the waste. Most non-high-level waste is destined for disposal when it leaves the nuclear facility. Nontransuranic wastes will probably be shipped to a commercial burial ground. Packaging and package contents must conform to requirements at the burial ground. (See Section 24.1.) Interim storage has been proposed for TRU wastes. (See Section 20.) Waste containers could be required to maintain their integrity in the storage environment for periods of 20 years or more. Some wastes may be shipped from their origin to a treatment facility. Although for the most part, these treatment processes are not now well-defined, the waste packaging used must be compatible with the head-end of the treatment process. For example, fiberboard or wooden boxes might be used for wastes being fed to a combustion-type treatment process. Additional considerations for selecting waste packaging are the convenience of the shipper and expense of the package.

22.7.4 Effects of Waste Treatment

Treatment processes in use or under development for non-high-level waste are generally designed to reduce waste volumes, recover transuranic elements and/or place the waste in a less hazardous physical or chemical form. A multitude of waste treatment processes are potentially applicable to non-high-level wastes. These processes are discussed in detail in Sections 8 through 10 and 12 through 14. The general effects of these treatments on transportation are outlined here.

- Incineration or chemical digestion processes convert the waste to an ash plus the contaminated residues of an off-gas treatment system. The major effect on transportation is to significantly reduce waste volumes and increase the specific activity of the waste. The latter could change shielding requirements.
- Mechanical compaction reduces waste volumes by about a factor of 5 with a corresponding increase in the activity per unit volume of the waste. Depending on the original waste and the shipping configuration, this could effect transportation shielding requirements.
- Solidification or immobilization processes tend to increase waste volumes and weight thereby decreasing specific activities. The solidification agent might also provide some shielding. These processes are proposed for both solid and liquid wastes.

- Several proposals have been made for processes to remove some of the TRU elements from TRU contaminated wastes. It is expected that the residue from these processes will still require disposal as TRU contaminated waste. (See Section 11.) The primary effect on transportation will be a reduction in waste volume and alteration of the physical and chemical form of the waste.

22.7.5 Available Transportation Systems

Many forms of non-high-level wastes have been shipped routinely in the United States for many years. These shipments have primarily been made by truck, although rail transportation has also been used. Available packaging is adequate for most anticipated forms of treated and untreated non-high-level waste. Because of the variation in physical form and radioactivity content of non-high-level waste, packaging will be described for general categories of waste.

- Low specific activity (LSA) (see Appendix E) material transported in sole use vehicles can be shipped in any strong, tight industrial packaging. Shipments of LSA material must still conform to the radiation dose requirements of 49 CFR 173.393. An example of shielded containers for LSA material is the BB-1 (Big Bertha) owned by Chem Nuclear Systems of Tacoma, WA. The BB-1 has a usable volume of 12.8 m³. Shielding can be varied from 2.5 to 5.0 cm of lead. The empty cask weight is about 15 tonnes. Similar shielded containers for LSA material are available from other companies.
- Type A quantities of non-high-level waste must be transported in packaging that meets the requirements of DOT Specification 7A. (See Appendix E). A wide variety of available shielded and unshielded containers meet these requirements. Most reactor wastes are presently placed in steel cylinders ranging in volume from 1.4 m³ to 5.7 m³ and shipped in shielded casks that meet the requirements for Type A packaging.⁽²³⁾ The steel cylinders are buried with the waste. DOT specification metal drums are widely used for shipments of radioactive waste that do not require radiation shielding. They are available in sizes ranging from 20 liters to 210 liters with maximum weights from 36 to 400 kg. Metal drums are relatively inexpensive and easy to load and handle. They are acceptable for burial and with appropriate closures and materials of construction could probably be made acceptable for TRU interim storage facilities. DOT specification wooden boxes and fiberboard drums are also used for transportation of Type A quantities of radioactive material. A list of DOT specification containers that have been determined to meet the requirements of Specification 7A is contained in Table 22.9. Other Type A waste packaging is also available. This includes steel bins in

TABLE 22.9. DOT Specification Containers Determined to Meet the Requirements of Spec. 7A(25)

<u>Container</u>	<u>Dot Specification Number</u>	<u>Container Capacity</u>	
		<u>Volume (liters)</u>	<u>Weight (kg)</u>
Steel Drums	6B	113	272
	6C	19	36
		38	72
	6J	210	400
	6L	Various	Various
	6M	Various	Various
	17C	19	45
		210	380
	17H	113	227
		210	380
Aluminum Drum	42B	210	250
Wooden Boxes	15A	4-230	9.5-152
	19A	43-230	66-181
	19B	43-230	34- 68
Fiberboard Drums	21C	57-210	27-181

NOTE: These containers are described in Reference 4 and in 49 CFR 178. Other DOT specification containers may also meet the Spec. 7A requirements. This list represents those that have been tested to date.

a variety of sizes, shielded casks of various volumes and shielding thicknesses, banded wooden boxes, concrete bins and others. Waste material requiring shielding is usually placed in a container that is easy to handle and meets burial ground requirements and shipped in a shielded overpack that is reused.

- Type B quantities of waste can be divided into two basic types--those requiring shielding to conform to the requirements of 49 CFR 173.393 and those for which no shielding is needed. Type B packaging is generally expensive to construct since it must meet the hypothetical accident test criteria of 10 CFR 71. Since most non-high-level waste is destined for disposal or long-term storage, it is economically attractive to package the waste in boxes or drums and ship these containers in a Type B overpack that can be reused. The boxes or drums used must meet the burial ground or storage facility requirements but do not have to qualify as radioactive material shipping containers.

There are several unshielded Type B overpacks available. The largest is the ATMX-600 railcar presently in use at ERDA's Rocky Flats facility. These cars have a 2.74 m x 2.74 m x 15.25 m cargo bay and a useful load of 45,800 kg. Up to 216-210-liter drums can be loaded directly into the car or 140 drums can be preloaded into two standard 2.43 m x 2.43 m x 6.09 m (8 ft x 8 ft x 20 ft) cargo containers. Plywood boxes are also used as waste containers. (The cars are described in detail in Reference 24.)

The "Super Tiger" manufactured by Protective Packaging, Inc., Louisville, KY is another large unshielded Type B overpack. It is a double steel walled container with 25 cm of polyurethane thermal and shock insulation separating the walls. Interior dimensions are 1.93 m x 1.93 m x 4.36 m and the outside dimensions conform to standards for 2.43 m x 2.43 m x 6.09 m (8 ft x 8 ft x 20 ft) cargo containers. The inner cavity will hold forty-two 210 liter steel drums. Maximum payload is 13,600 kg and the containers can be transported by truck, rail, or cargo vessel.⁽¹¹⁾ This container is shown in Figure 22.13.

There are several other Type B overpacks available. The NECO B2 overpack has a capacity of fifteen 0.21 m³ steel drums. The Protective Packaging

FIGURE 22.13. Protective Packaging, Inc., "Super Tiger" Type B Overpack (Reference 11)

"Paper Tiger" is designed for a single 0.21 m^3 drum and their Poly Panther accommodates a special $1.28 \text{ m} \times 1.48 \text{ m} \times 1.83 \text{ m}$ steel bin.

Shielded Type B overpacks are typically smaller than unshielded containers. The container volume must be limited so that shielded weight does not cause overall package weights to exceed truck or rail weight limits. Shielded Type B packaging is available in a variety of sizes with various shielding thicknesses. The NECO B3-1, for example, provides 15 cm of lead shielding and holds a single 210-liter steel drum. The Brookhaven Waste Shipment Overpack⁽¹¹⁾ is a combination concrete vault and steel overpack. Two containers are available with volumes of 0.32 m^3 and 1.2 m^3 . The ACTOR LL-50-100, LL-57-65 and BC-48-220 containers have volumes of 3.71 m^3 , 1.93 m^3 , and 6.04 m^3 respectively. Table 22.8 gives a partial list of available shipping casks that provide relatively large amounts of shielding. Such casks might be used, for example, to transport highly radioactive HEPA filters or failed equipment from reprocessing plants. There is also the possibility of adding shielding to the unshielded overpacks described above, although this can severely limit payload weights.

The list of Type B containers discussed here is by no means comprehensive. Many other containers exist. The containers discussed do, however, illustrate the status of technology. Packaging is available for transporting the wide range of material included in the category of non-high-level waste.

22.7.6 Transportation Systems Under Development

Available technology is adequate to meet the transportation requirements for low and intermediate level wastes. Most of these wastes have been packaged and shipped routinely for many years. Future developments in the field of packaging for non-high-level waste will be primarily aimed at handling the increased waste volumes that will be produced as the LWR industry expands. New containers will be developed to more efficiently handle a particular waste form, but their design and construction will be basically an application of existing technology.

Developing containers for waste forms that have not been transported extensively in the past (e.g., gases) will require longer lead times, but will still be basically an application of existing technology. Containers discussed previously can be used to transport tritium and iodine chemically fixed on solids. A ^{85}Kr shipping container has been developed. This container is pictured in Figure 22.14. Because of heat removal considerations, a maximum of about 74,000 Ci of ^{85}Kr can be shipped in the cask at one time. The gas cylinder pressure is about 34 atm and the container provides about 5 cm of lead shielding. A conceptual design study has been carried out for noble gas shipping containers that could be used to ship the radioactive noble gases from an operating power plant.⁽²⁶⁾ This design assumes that the gases will be

FIGURE 22.14. ^{85}Kr Shipping Container

collected in a 50-liter capacity cylinder at a pressure of 150 atm. The filled cylinder would be placed in a water-filled cask weighing about 6.4 MT. The water acts as shielding and a heat transfer media. Six of these casks could be shipped to a Federal repository on one railcar.

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